Supercritical-water-cooled reactor system (SCWR) System Safety Assessment

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1. Summary

Conceptual Super-Critical Water-cooled Reactor (SCWR) designs have been completed in Canada, EU and Japan. The thermodynamic expression "supercritical" describes the state of the substance where there is no clear distinction between the liquid and the gaseous phases. The SCWR design was one among six different Generation IV system designs selected on a fourteen-country initiative expressing a strong interest in collaborative Research and Development (R&D) to develop future generation of nuclear energy systems for deployment beyond 2030.

The SCWR concepts follow two main types, namely the reactor pressure vessel (PV) analogous to conventional Light Water Reactors (LWRs), and distributed pressure tubes (PTs) or pressure channels analogous to conventional Heavy Water Reactors (HWRs). The safety analysis of the Canadian SCWR concept and the EU High Performance Light Water Reactor concept covered key accident scenarios expected during operation. The large-break loss-of-coolant accident (LOCA) is assumed to be the limiting scenario for the Canadian SCWR with a maximum predicted cladding temperature of 1075°C. A limiting accident scenario is defined as a scenario that gives rise to maximum cladding temperature. The peak cladding temperature calculated for an HPLWR accident, initiated by an inadvertent isolation of all main feedwater and main steam valves with a delayed low-pressure injection to the core, is about 910°C. On the other hand, the limiting accident scenario for HPLWR is the small break (15%) loss-of-coolant accident, where the peak cladding temperature is about 1000°C (350°C above the steady-state temperature).

The safety assessments concluded that there are no major impediments to further developing SCWR design concepts. However, there are gaps in understanding how the reactor core will behave under SCWR pressures and temperatures. The proposed core concepts require a number of engineering assessments to evaluate the structural integrity and heat transfer characteristics. The results of these analyses will provide feedback to further refine the core design concepts.

2. General overview of the performance goals

In 2001, nine countries, including Canada, initiated the Generation IV International Forum (GIF) to collaboratively undertake R&D on the next generation of nuclear energy systems [1]. Today, eight GIF Charter signatories, as well as Euratom, are actively participating in the GIF, through the Framework Agreement. The GIF Charter provides a general framework for GIF activities and outlines its organizational structure. The GIF defined goals in four broad areas of: Safety and Reliability; Sustainability; Economics; and Proliferation Resistance and Physical Protection (PRPP) in the original Technology Roadmap [1] and also in the updated [2] version. Super-Critical Water-cooled Reactor (SCWR) systems aim at meeting or exceeding the current fleet of nuclear reactor systems in these four areas.

From a safety point of view, the SCWR systems adopted multiple passive safety systems, in complement to active systems that enhance their safety characteristics compared to the current fleet of nuclear reactor systems. The design of the safety systems adheres to the "Defence-in-Depth" safety principle. The performance goals in developing the SCWR systems are to control criticality of the reactor at all times, specifically during normal operation and anticipated operation occurrences; cool the fuel and the core; and contain release of radioactivity.

There are two main designs of SCWR; one that uses a Pressure Vessel (PV) and the other uses Pressure Tubes (PT). A comparison of operating conditions of Boiling Water Reactor (BWR), CANDU Pressurized Heavy Water Reactor (PHWR), and the Pressurized Water Reactor (PWR) are shown in Figure 1 along with the two SCWR designs [3]. Lines defining the critical pressure and critical temperature confine the boundaries of liquid and vapour phases, respectively. The region beyond these boundaries represents the supercritical water. SCWRs are designed to operate at 25 MPa, which is higher than the critical pressure for water (22.1 MPa). The pressure vessel SCWR is designed to operate at steam temperatures between 280 and 500°C, while the pressure tube design will operate between 350 and

625°C. A comparison of SCWR design parameters proposed by six different organisations are shown in Table 1. Among the six designs, in terms of pressure and temperature, the Canadian SCWR and the European High Performance Light Water Reactor (HPLWR) can be considered two distinct types. In this document, only these two reactor systems will be discussed as representative reactor types for the SCWR concept

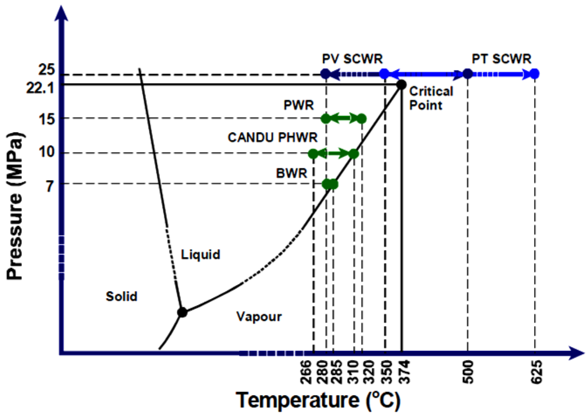

Figure 1: Operating Conditions of Reactor Designs [3]

3. Sustainability

One of the primary GIF goals is enhanced sustainability in reducing waste associated with fuel cycles and conservation of natural uranium reserves. The clean air objectives of Gen IV sustainability goals are promoting long-term availability of systems, helping effective fuel utilization, minimizing nuclear waste, reducing the long-term stewardship burden of nuclear waste, and improving protection for public health and the environment.

Table 1: SCWR Design Parameters of Proposed Reactor Concepts [3]

Parameters	Unit	Canadian SCWR	SCWR-M	HPLWR	JSCWR	Super Fast Reactor	VVER-SCP
Country	–	Canada	China	EU	Japan		Russia
Organization	–	CNL	SJTU	EU-JRC	Japanese Consortium	University of Tokyo	OKB "Gidropress", IPPE
Reactor type	–	PT	RPV	RPV	RPV	RPV	RPV
Spectrum	–	Thermal	Mixed	Thermal	Thermal	Fast	Fast-resonance
Power thermal	**MW_{th}**	2540	3800	2300	4039	1602	3830
Linear heat rate max/av.	**kW/m**	31/25	39/18	35/14, 8, 4.5 (a)	-/13.5		-/15.6
Gross thermal efficiency	**%**	49.4	～44	43.5	42.7	~44	43-45
Pressure	**MPa**	25	25	25	25	25	24.5
T_{in} coolant	**°C**	350	280	280	290	280	290
T_{out} coolant	**°C**	625	510	500	510	508	540
Flow rate	**kg/s**	1263	1927	1179	2105	820	1890
Active core height	**M**	5.0	4.5	4.2	4.2	2	4.05
Equiv. core diameter	**M**	~5.5	3.4	3.8	3.34	1.86	3.6
Fuel	–	Pu-Th	UO_2/MOX	UO_2	UO_2	MOX	MOX
Cladding material	–	Alloy 800H	SS	316SS	310SS	SS	austenitic alloy (ChS-68, EP-172)
Number of Fuel Assemblies		336	284	1404	372	162/73	241
Number of Fuel Rods in Fuel Assemblies		64	180/324	40	192	252/127	252
D_{rod}	**mm**	9.5/10.0 (b)	8	8	7	5.5	10.7
Pitch	**mm**	vary	9.6/9.6	9.44		6.55	12
T_{max} cladding	**°C**	850	Not specified	620	700	643	Not specified
Moderator	–	D_2O	H_2O/---	H_2O	H_2O	-/ZrH	H_2O

(a) Evaporator, Super heater 1, Super heater 2; (b) Inner and outer rings

The Canadian SCWR concept adopts an advanced thorium-based fuel cycle using an oxide mixture of thorium with reactor-grade plutonium, recovered from used LWR fuel. This alternative, rather than the use of low enriched uranium (LEU), extends the lifetime of natural uranium (NU) reserves. Thorium is estimated to be three to four times more abundant than uranium in the world [4], providing a long-term fuel supply. Furthermore, the known thorium reserves are more evenly distributed over wide regions of the planet, compared to known uranium reserves [5]. Therefore, more countries can utilize nuclear power through the implementation of thorium fuel cycles. The neutron economy associated with the use of a heavy-water moderator also provides some advantage over comparable light-water moderated reactor cores, with respect to fissile utilization, the energy yield per unit mass of fissile material.

The European High Performance Light Water Reactor (HPLWR) [6] envisages using UO_2 fuel with ^{235}U-enrichment at approximately 7% [7], which is higher than the current Pressurized Water Reactor (PWR) levels. This helps effective fuel utilization, however, an improved cladding material is being used to achieve higher enrichment. In addition, the SCWR is considered to be a more evolutionary concept since it is based on water-cooled thermal-spectrum reactors such as PWRs, BWRs and PT-HWRs that are in widespread use today.

The thermal efficiency of a Canadian SCWR is 49%, compared to 34% from LWR and PHWR, which is an approximately 40% increase in relative thermal efficiency. This provides an added advantage to sustainability through enhanced resource utilization and minimization of waste.

Evaluations of the impact of the Canadian SCWR concept on high level waste focused mainly on determinations of decay heat and radiotoxicity. It has been shown in [8] that the reuse of plutonium from used HWR fuel or LWR fuel, in the Canadian SCWR concept, can result in significant reductions (between 25% and 50%) in decay heat and radiotoxicity for long term storage (i.e., from 1,000 s to 10,000 s of years). In reference [8], it was also shown that the high level waste per unit electricity produced was also reduced (by approximately 25%) when HWR Pu is recycled in the Canadian SCWR concept.

3.1 Economics

The SCWR concept adopts a direct thermodynamic cycle at pressure and temperature matching closely the current advanced turbine configuration of Super-Critical Water (SCW) fossil fuel fired power plants. The direct cycle eliminates the need for steam generators (as in PWRs) and the single phase supercritical fluid eliminates the need for steam separators (as in BWRs). This simplifies the plant concept, leading to cost reductions compared to the current Water Cooled Reactors (WCR) [9].

Three main benefits from plant-cycle efficiency improvements are:

- Increasing the power output for the same fuel input (specific fuel utilization),
- Reducing waste heat from turbines and condensers (environmental discharges), and
- Building fewer plants to meet demand (capital and operating cost savings).

The economic analysis of the Canadian SCWR concept assessed it against the GIF enhanced economics metric. GIF uses a standardized method to evaluate the economics of Gen-IV nuclear energy systems [10]. The GIF Economics Modelling Working Group developed an Excel-based model [11]. The model and associated cost estimating guidelines [11] were used to evaluate the economics of the Canadian SCWR concept for two fuel options, and compare the results to the economics of the Advanced Boiling Water Reactor (ABWR). The ABWR was chosen for comparison because of similarities in components and systems, and its use in other Gen-IV economic analyses, specifically the European HPLWR.

Figure 2 shows the range of Total Capital Investment Cost (TCIC) possible for each nuclear technology established from the uncertainty in the cost estimates. Based on the mid-point TCIC (represented by an “x”), where the cumulative probability is 50%, the Canadian SCWR concept scenarios are similar to the ABWR. In fact, the total capital cost for the Canadian SCWR concept base case estimate is ~$260 M less than the ABWR estimate. However, because the ABWR has a net installed electrical capacity of 1371 MWe, 194 MWe more than the Canadian SCWR concept, the ABWR's TCIC is slightly lower.

As illustrated in Figure 2, both SCWR concept scenarios have a wider TCIC 80% confidence interval (grey boxes), and a wider total range (black lines), than the ABWR. This is expected when comparing a new reactor concept with a mature reactor technology that has already reached the nth-of-a-kind (NOAK) stage.

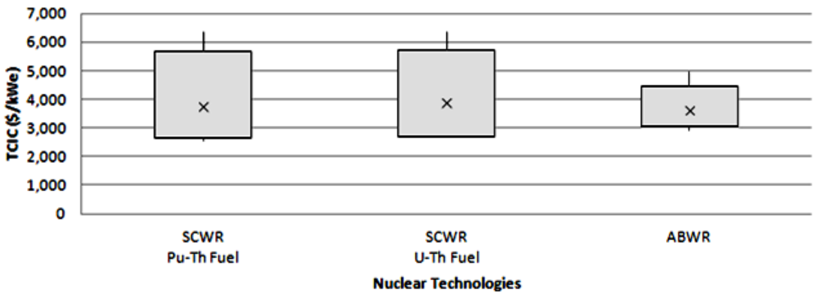

Figure 2: Comparison of Nuclear Technologies by Total Capital Investment Cost

The electricity generation costs of a HPLWR with a typical LWR was compared [6] using the Light Water Reactor (LWR) capital cost of 3 000$/kWe obtained from the literature [12] and converted to 2 200€/kWe. Using the GIF Cost Estimating Guidelines [11], cost break down was estimated for both the 1 000 MW Advanced Boiling Water Reactor (ABWR) reference plant and the HPLWR. Assuming common structures and component cost, the design details available thus far for HPLWR were evaluated for cost reductions. As shown in Figure 3, about 20% cost reduction can be expected in all of the major cost categories assessed. For example, the size reduction of the reactor equipment such as reactor vessel, control rod drive system, reactor internals, main heat transport system and safety systems will reduce the construction costs by about 41% (equivalent to about M€ 80). In structures and improvements in containment structure and reactor auxiliary building will save about 22.4% (M€ 96) compared with the reference value. The specific plant construction costs is 1,795 €/kWe for the HPLWR and 2,255 €/kWe for the reference plant. All of these estimates have a high uncertainty and as more information on the HPLWR design becomes available, the numbers will be re-evaluated.

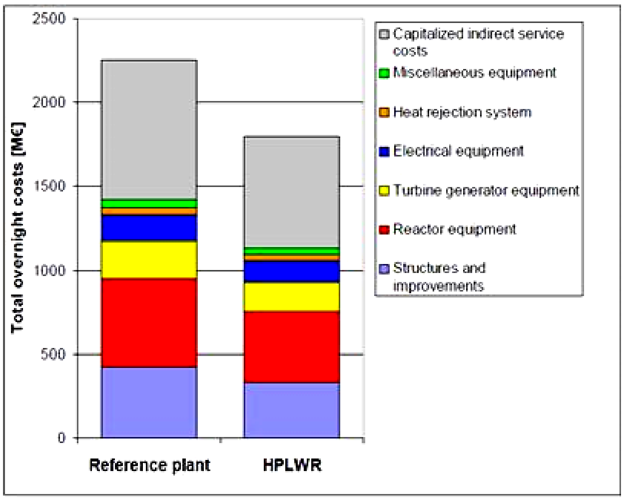

Figure 3: Estimated Cost of the HPLWR Power Plant in Comparison with a Scaled ABWR as Reference Plant

3.2 Safety and reliability

The European HPLWR and the Canadian SCWR designs use a once through steam cycle, in which steam from the core outlet is directly supplied to the high pressure turbines. This feature is similar to boiling water reactors (BWR) and therefore similar safety systems can be used. The HPLWR, however, adopted a different coolant flow path inside the reactor, where the complications of using separators and dryers are avoided unlike in a BWR. In the HPLWR design shown in Figure 4, the control rod drives are on the outside and the control rod guide tubes are inside the reactor, which can be adopted from PWR design without significant modifications. The decay heat removal in the HPLWR, following simultaneous feed water pump trip, is achieved through forced convection inside the reactor as shown in Figure 5. The reactor is automatically tripped and depressurized following the loss of feed water pumps as in a station blackout. The depressurization occurs through a steam turbine, driving a high pressure coolant injection pump, providing forced cooling.

A passive system without rotating components is also provided in the HPLWR using steam condensation in an additional upper pool inside the containment (Figure 6). Since this system would likely lead to slow reduction in system pressure, an innovative system with a steam injector has been added. In this method, following a short initial depressurization through the Automatic Depressurization System (ADS), subcooled liquid from the upper pool is sucked into the steam injector, building sufficient condensate pressure to refill the vessel with condensate. Additional details of this mechanism is discussed in Section 6.3.

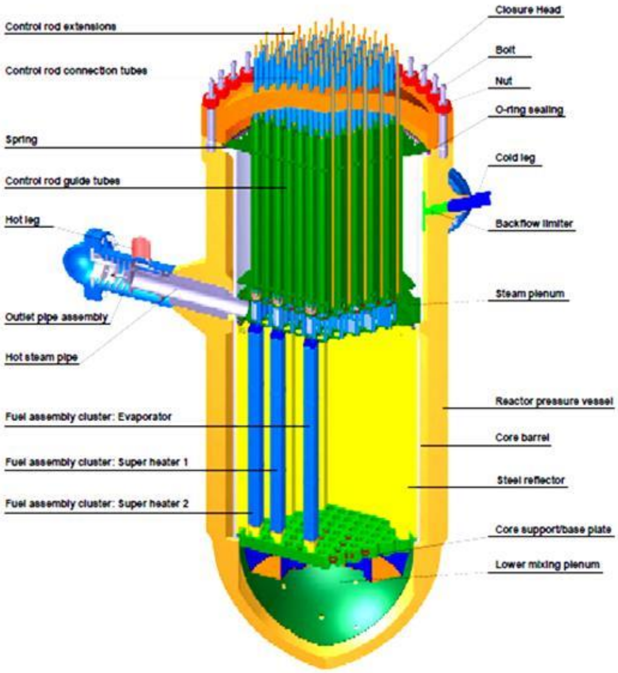

Figure 4: HPLWR Pressure Vessel and Internals [6]

The Canadian SCWR concept shown in Figure 7 contains passive safety features to enhance its safety characteristics. In addition to these passive safety features, the implementation of the insulated fuel assembly facilitates continuous cooling of the fuel through passive heat rejection through the insulator into the moderator in the event of a LOCA without emergency core cooling (Figure 8). The heat in the fuel is transferred by radiation to the liner tube and conducted through the insulator and the pressure tube to the moderator [13]. Together with the passive moderator cooling system, the decay heat from the fuel can be transferred to the ultimate heat sink continuously, even without operator intervention, maintaining the fuel cladding below its melting point. The effectiveness of the passive moderator cooling system has been verified experimentally in a full-height, reduced volume, test facility [13]. The safety analysis predicted relatively low peak cladding temperatures for various accident scenarios. This confirms the potential of meeting the “no-core-melt” goal for the Canadian SCWR concept.

Thorium-based fuel has a much higher melting point and thermal conductivity than uranium oxide fuel. This feature enhances the safety characteristics of the fuel. Mixing thorium with plutonium in the form of $(Th,Pu)O_2$, however, may have a negative impact on these characteristics. Available experimental data on thermal conductivity for the $(Th,Pu)O_2$ fuel are scarce and large scatter was observed among the limited data. Additional experimental data on thermal conductivity are required for the $(Th,Pu)O_2$ fuel to confirm the fuel temperature predictions. In addition, the effect of changes to gap conductance, based on grain boundary inventory release experiments, are required. The cladding material selected is an Oxide Dispersion Strengthened material called PM2000. This is a highly oxidation resistant and extremely creep resistant ferritic iron-chromium based alloy. The alloy has 74.45 wt% Fe, 19 wt% Cr, 5.5 wt% Al, 0.5 wt% Ti, 0.05 wt% C, and 0.5 wt% Y_2O_3.

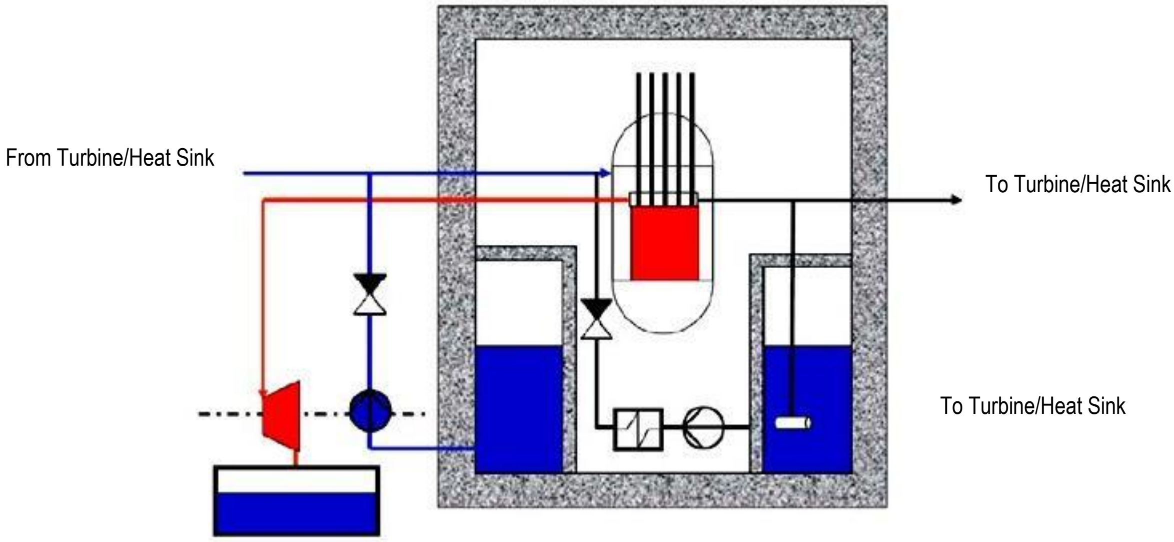

Figure 5: Steam Turbine Driven High Pressure Coolant Injection Pump During HPLWR Automatic Depressurization

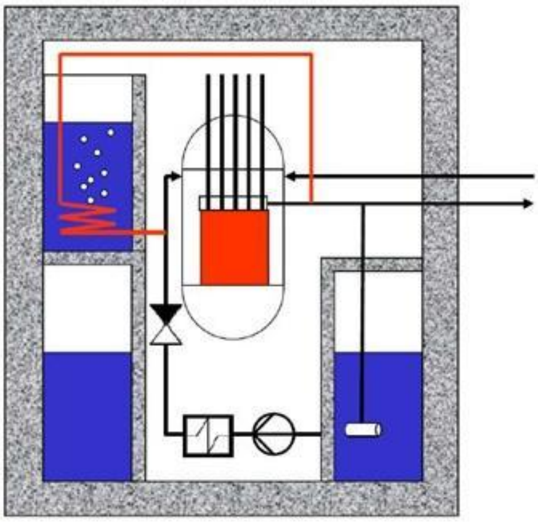

Figure 6: Condensation in an Upper Containment Pool in an HPLWR Closed Loop Depressurization

In compliance to GIF Integrated Safety Assessment Methodology (ISAM) [14], preliminary analyses using the ISAM tools such as Qualitative Safety Review (QSR) and Phenomena Identification and Ranking Tables (PIRT) were applied during the concept development phase of the Canadian SCWR. Although a Probabilistic Safety Assessment (PSA) is not required at the conceptual development phase, a simplified PSA has been performed to quantify the core damage frequency (CDF) for the safety system of the Canadian SCWR concept. The ISAM application to SCWR the concept was summarised in a white paper [15].

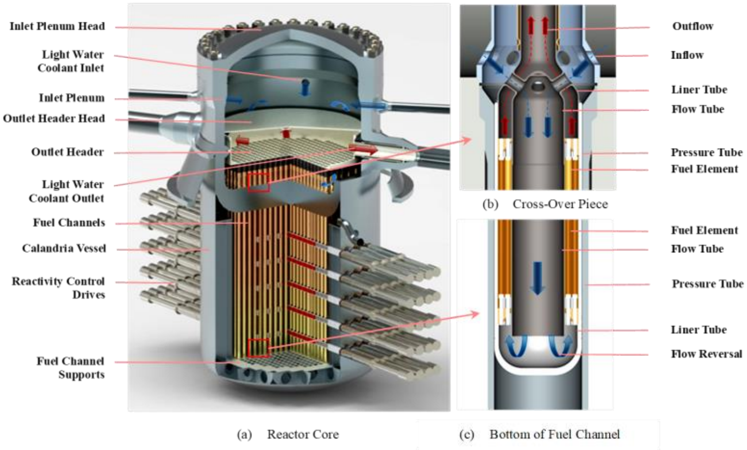

Figure 7: Canadian SCWR Core Concept

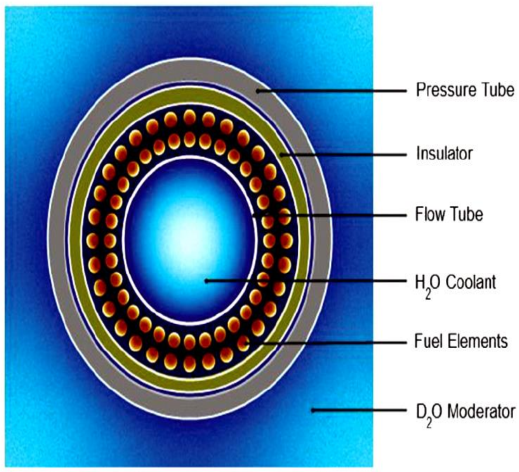

Figure 8: Cross-Sectional View of Canadian SCWR Fuel and Fuel Channel Concept

3.3 Proliferation resistance and physical protection

Proliferation resistance and physical protection is one of the four central goals for the development of Gen IV reactor concepts. The Gen IV nuclear energy systems are expected to be unattractive and the least desirable route for diversion or theft of weapons-usable materials [1], and will provide increased physical protection against acts of terrorism.

A PRPP methodology [16] was created by the GIF PRPP working group in order to evaluate the proliferation resistance and physical protection of Gen IV nuclear energy systems. There are a number of groups that may attempt to make nuclear weapons; the PRPP methodology identifies potential "actors" as either proliferant states or sub-national adversaries [16].

The Canadian SCWR concept adopts an advanced thorium-based fuel cycle to enhance the ability to safeguard nuclear materials. Thorium itself is a non-fissile, fertile material. However, it requires a fissile material to start the process. The thorium fuel produces mainly ^{233}U and no plutonium than natural or enriched uranium fuel. Therefore, it is somewhat less attractive to be diverted for weapons. In addition to ^{233}U, the thorium fuel also produces ^{232}U as a by-product. The ^{232}U has a high gamma-emitting decay chain and requires special handling [4]; the presence of ^{232}U with ^{233}U is a form of self-protection against easy and safe retrieval of ^{233}U. This is unlikely to be a deterrent for groups who are not concerned with the hazards associated with high gamma-emitters. It is also difficult to separate ^{232}U from ^{233}U, therefore making it difficult to extract the fissile material.

There are relatively few materials that may be considered proliferation targets in the Canadian SCWR nuclear energy system. The fresh fuel in a SCWR has low quality plutonium in terms of host state preference for weapons manufacture. Concealed production of material in the Canadian SCWR concept, while possible, is not easily achieved. The reasons are that the batch-fuelled core allowing easy monitoring and verification of fuel inventories. It is expected that safeguards design and inspection/monitoring of fuel inventories will build upon the current, proven approaches for LWRs thus resulting in high detection probability. All fertile and fissile material present in the system will be contained within either fresh or used fuel. Fresh fuel assemblies are expected to be transported to the reactor site, stored on-site prior to being loaded into the reactor, irradiated for three 395-day cycles, and discharged into used fuel wet shielding and cooling. Since one third of the core is replaced at each refuelling outage, there will necessarily be at least one refuelling batch of 112 fresh fuel assemblies, held within fresh fuel storage prior to refuelling. The fresh fuel storage concept is developed to store up to 150 fuel assemblies. The fuel transfer pool, located adjacent to the refuelling well, is used to temporarily store the spent and new fuel assemblies before transferring them to the reactor vessel or the spent fuel pool.

Based on recent calculations, the fuel will remain in the core for three cycles, totalling approximately 3.2 full-power years. The spent fuel pool is located in the reactor auxiliary building. It can store 1344 fuel assemblies to accommodate used fuel assemblies for ten years plus one full core. After ten years, the fuel is assumed to be transferred to dry storage on site, though this has not yet been determined in the concept.

The main threats identified for the Canadian SCWR concept are the proliferation resistance threats such as theft or diversion of fresh and/or spent fuel, and concealed production of material in the reactor (i.e., misuse of reactor), and physical protection threats such as sabotage attempts to cause radiological release. The risks involved with sabotage are expected to be similar to existing LWRs and HWRs, and are not discussed further here.

Concealed production of material in the SCWR concept such as in HPLWR, while possible, is not easily achieved. The batch-fuelled core allows easy monitoring, improves the ability to safeguard, and the ability to easily verify fuel inventories against production records [6]. Based on the similar nature of the once-through fuel cycle in existing reactors, it is expected that safeguards design and inspection/monitoring of fuel inventories will build upon the current, proven approaches for LWRs, thus resulting in high detection probability. Based on its similarity to existing light water reactors, the SCWR concept should be able to use safeguards methods based on decades of experience of batch-fuelled reactor technology.

4. Historical Review of, and Feedback from, Past Construction and Operation Experiences

The use of supercritical water to increase thermal efficiency and avoid phase separation in the thermodynamic cycle was adopted in coal-fired power plants. The term supercritical is a thermodynamic expression that describes the state of the substance where there is no clear distinction between the liquid and the gaseous phases. The world's first high-pressure (17.5 MPa) and high-temperature (610°C) power plant with a once-through boiler started operating in Leverkusen, Germany in 1949. Following this, in 1954, the world's first steam turbine with supercritical steam conditions (30 MPa and 600°C) started operating.

The use of fossil-fuel-fired supercritical water power plants (FFSP) is the largest application of fluid at supercritical pressures. Temperatures at the boiler exit were initially around 550°C, but recent advances in materials and turbine technology have led to units using 625°C at 25 MPa, with R&D proceeding towards adopting outlet temperatures of over 750°C at a pressure of 35 MPa [17]. The

Ultra Super Critical turbine manufacturers now claim [18] that developments in FFSPs will lead to thermal cycle efficiencies of greater than 50%.

The SCWR concept utilizes proven technical advancements gained in coal-fired power plants by increasing the live steam temperature to improve thermal efficiency and specific turbine power. The fossil fired power plants with supercritical steam have operated for over 60 years; this success can be attained in nuclear power plant concepts by incorporating proven design of turbines, feed water pumps and other components of the steam cycle. In parallel, the containment design for the SCWRs can be derived from the latest boiling water reactors (BWR) and therefore the research and development needs can concentrate on the reactor-core design.

4.1 Brief history

During the first commercial water-cooled reactor development period, many countries evaluated coolant at supercritical temperatures and pressures [19]. The idea of using SCW as the coolant in a water-cooled reactor dates back to the 1950s and 1960s [19] - [23]. Although reactors operating at supercritical temperatures and sub-critical pressures (superheated steam) were tested, no reactor operating with a coolant above the thermodynamic critical point of water (373°C, 22.1 MPa) was ever built. The Russians developed significant experience with the operation of nuclear steam reheat channels in the pressure-tube BWRs at the Beloyarsk NPP; these nuclear steam reheat plants operate at temperatures above the critical point of water, but at subcritical pressures. Two units at Beloyarsk employed boiling channels and superheat channels. The boiling channels in Unit 1 were part of a closed loop that exchanged heat through a steam generator to produce saturated steam for the superheat channels, while Unit 2 was a direct cycle [23].

During the intervening years, major advances have been made in the fossil power and thermal plant industry using SCW, and there is now a potential to significantly improve nuclear plant cycle efficiency with SCW. Reaching 50% efficiency can yield over 40% more electrical output for the same fuel input compared to current LWR and HWR designs. In addition, the waste heat rejection is reduced by 30% and there are opportunities for co-generation. In general, the total thermal efficiency of a modern thermal power plant using subcritical steam is about 36–38% whereas the use of supercritical steam increases the efficiency to about 45-50%. If ultra-supercritical steam is used, the efficiency can be improved to 50% or more. The highest total thermal efficiency achieved in today's thermal power industry is about 56–58% with a combined cycle of gas turbine – steam turbine. To improve the cycle efficiency to 45-50%, the pressure tube type SCWRs would operate at pressures at or near 25 MPa and core outlet temperatures up to 625°C.

While no supercritical water cooled reactor has ever been built, there has been no scarcity of SCWR designs. In 1957, a light-water-moderated, supercritical steam cooled reactor was designed by Westinghouse. In this design, fuel assemblies with 7 fuel rods in cylindrical, double walled cans were used to insulate the superheated steam from the liquid moderator water at 260°C. To avoid activity in the turbine, an indirect steam cycle was adopted. In 1959, a 300 MW thermal heavy water moderated and light water cooled reactor was designed by General Electric with once-through steam cycle. The coolant reached 621°C after passing through the core four times in the design. In 1962, Westinghouse designed a 1000 MW, graphite moderated and light water cooled, pressure tube reactor and was named the Supercritical Once Through Tube Reactor (SCOTT-R). The thermal efficiency of the design was 43.5%. The SCWR concept received renewed interest in Japan when scientists and students at the University of Tokyo developed cores with thermal neutron and fast neutron spectrums [19]. All of the recent design concepts and their analyses are documented by Oka et al. [19].

4.2 Feedback from Past Construction and Operation Experience

The SCW operating experience for a nuclear power plant stems from a BWR that used an integral nuclear superheater [20]. This reactor was built in Grosswelzheim, Germany, and provides an evolutionary step from boiling water reactors to a SCWR. The 100 MW thermal power prototype was built between 1965 and 1969 reaching criticality in October 1969. About 75 MW thermal power was used in evaporating the coolant between the fuel rods like in a conventional BWR, and another 25 MW was used in the superheater to reach supercritical conditions unsuccessfully. The plant operated with a reduced temperature of 457°C of superheated steam at 9.1 MPa reactor inlet pressure as an introductory step. Following commissioning, the superheater failed and the reactor operation was terminated.

A limited amount of lessons learned from the past experiences have been incorporated in conceptual designs. The operating experience from existing boiling water reactors and the material behaviour experience from ultrasupercritical steam applications in the fossil energy boilers have been used in the conceptual design. The available operating experience is scarce on radiological effects on supercritical water. Currently several studies are attempting to improve understanding in this area.

5. Level of Ongoing Safety-Related Research and Development

Safety-related R&D projects are being performed by members of the GIF SCWR Thermal-Hydraulics and Safety Project Management Board and to a lesser extent by the Materials and Chemistry Project Management Board in support of the development of SCWR. Most of the projects have been introduced to expand the fundamental science required for examining heat transfer characteristics during postulated accident scenarios (such as power and flow transients), the effectiveness of the liquid injection shut-down system, no-core-melt behaviour, and critical flow behaviours, in support of pressure relief valve design and analyses of large-break loss-of-coolant accidents. Other projects are extending the application of safety analysis codes and developing the tools for event-tree analyses of core damage frequency during postulated accident scenarios.

The thermal hydraulics and safety R&D activities are progressing through national programs in Canada, China, Europe and Japan. The emphasis is to provide relevant experimental data for verification and validation of analytical toolsets used in design and in safety analysis, with an objective of improving the accuracy of prediction. Another active area of R&D is optimising fuel assembly design and obtaining heat transfer data with annuli, 3-rod, and 4-rod assemblies. Additional R&D is focused on heat transfer in subchannels of SCWR fuel assemblies, and the database of heat transfer correlations. Previous assessments of heat transfer correlations, against an extensive database of heat transfer in tubes at supercritical pressures, revealed deficiencies in predicting the heat transfer coefficients accurately over the range of bulk fluid temperatures of interest to SCWR analyses. Additional experimental heat transfer coefficients are being generated to improve prediction accuracy.

In European experimental facilities, supercritical Freon is being used as working fluid to understand heat transfer deterioration under SCWR conditions. The main purpose of some of these experiments is the study of mixing behaviour of supercritical fluids under strong density variation and buoyancy effects.

Experimental studies in China concentrate on safety-performance-related tests including: natural circulation; critical flow; Critical Heat Flux near critical pressure; flow stability in parallel channels; assessment and applicability of CFD codes; and research on scaling methods of different supercritical fluids. Candidate internal structural materials and fuel cladding materials will undergo irradiation testing in a newly-designed high-temperature material irradiation test apparatus in China. The material irradiation data from these tests and the corresponding results of post-irradiation examination will be an essential part of SCWR R&D. An in-pile fuel assembly irradiation test loop, with supercritical water in a research reactor, is planned to simulate the typical operating conditions of the SCWR fuel assembly and conduct the irradiation test to qualify its performance.

A new benchmark exercise has been launched to evaluate simulation tools modelling supercritical water flow through a 2×2 rod bundle. The experimental data will be provided by Nuclear Power Institute of China (NPIC) and Canadian Nuclear Laboratories (CNL). The participants will first perform a blind-calculation based on the operating conditions and geometry parameters, and subsequently compare it with the experimental data.

A Round Robin corrosion exercise, involving 12 partners from the EU, Canada and China, is to compare the results of corrosion tests in different test facilities; the main objective is to identify the origins of differences observed in the results of a previous Round Robin exercise. The parameters suspected to be the cause of the differences are: coupon preparation, differences in flow and mass transfer rates.

The Research Centre Řež (Czech Republic) completed a number of out-of-pile supercritical water loop commissioning tests in 2017. Several future R&D activities are being planned in this facility. Some of the activities are to qualify new designs of fuel and testing safety critical components.

Oxidation testing of samples of SS310S and SS316L, at 500°C in SCW, is underway for prolonged exposure times that exceed the proposed in-service life of the Canadian SCWR fuel cladding, and these data will be extremely important in demonstrating the long-term performance of these materials. Molecular dynamics simulations are being performed to understand the thermochemical processes at the steel surfaces, with higher concentrations of reactants such as oxygen than in the bulk fluid. Examination of several years of corrosion and supercritical water exposure data for alloy 625 and alloy 800 is underway.

6. Achievement of Fundamental Safety Function

Targets for the development of the Canadian SCWR core physics concept are based on the overall goals for GEN-IV reactor concepts: advanced safety, enhanced sustainability, enhanced economics and advanced non-proliferation [24]. Advanced safety is incorporated through negative reactivity coefficients.

6.1 Reactivity Control

The reactivity of a reactor is controlled by a set of mechanisms used for regulation (i.e., control) and protection (i.e., safety). These reactivity control mechanisms make changes to neutron absorption and thus control reactor power. For continuous short-term reactivity control only a few mk of reactivity is necessary. Reactivity controls and protection systems function throughout the design life to preserve the integrity of the fuel and the core under all expected conditions of normal operation and anticipated operational occurrences, with appropriate margins for uncertainties. The purpose of the reactor regulation system is to: maintain the neutron multiplication constant, k, equal to one for steady operation; provide small negative or positive changes in Δk; and prevent flux oscillations. The reactor protection system, on the other hand, is designed for the rapid insertion of a large amount of negative reactivity to shut down the reactor. Localized flux changes may occur in the core during refuelling or due to other factors. These oscillating unbalanced reactivity loads are counterbalanced in various regions called zones of the core. A zone control system is used to maintain the flux to optimal levels.

6.1.1 Control (and shutdown) systems

Reactivity suppression for the Canadian SCWR concept is achieved through the use of burnable neutron absorbers. The integral fuel burnable absorbers (IFBA) are an attractive option for reactivity hold-down because they circumvent the potential positive increase in Moderator Temperature Reactivity Coefficient (MTC) resulting from addition of soluble poisons to the moderator. The MTC can become positive only when gadolinium is used as a soluble absorber for reactivity management. An increase to moderator temperature decreases both the density of the moderator and the density of gadolinium. As a result, with excess reactivity from a batch of fresh fuel, and the use of a soluble absorber like gadolinium nitrate in the moderator to suppress reactivity, the MTC can become positive. Potential IFBA include ZrB_2 [25] and the rare earth oxides, Gd_2O_3, Sm_2O_3, Dy_2O_3 and Er_2O_3 [26]. Since ZrB_2 cannot be mixed directly with fuel, it is incorporated with fuel as a coating on the fuel pellet surface. ZrB_2 has the advantage that its rate of depletion is well suited for typical cycle lengths (i.e., it can be completely depleted by the end of one cycle), but disadvantages include a reduction of heat transfer from the fuel to the fuel cladding and an increase in internal pressure from the production of helium. Of the potential rare earth IFBA, Gd_2O_3 and Er_2O_3 were found, based on the calculation results presented in [26], to be the best options for reactivity control in PWRs. For the Canadian SCWR concept, Gd_2O_3 and Er_2O_3 were chosen as potential IFBA because of prior experience with PWRs and BWRs. ZrB_2 is not under consideration because of its negative impact on heat transfer and the internal pressurization associated with helium production.

For reactivity hold-down, IFBA can be uniformly incorporated into all the pins of specific fuel assemblies or all the fuel assemblies in a fuel batch. For power levelling, IFBA can be selectively incorporated into specific fuel pins of specific fuel assemblies and/or in axial regions or zones.

WIMS-AECL infinite-lattice burnup calculations were performed as a first step in the comparison of Er_2O_3 and Gd_2O_3. In these calculations, Burnable Neutron Absorber (BNA) was uniformly incorporated in the fuel and incorporated in all of the fuel pins. The purpose of the burnup calculations was to determine the impact of a neutron absorber on the initial excess reactivity of the fuel, the extent of reactivity swing when the neutron absorber is depleted, and the overall impact on the excess reactivity as a function of burnup.

In PWRs and BWRs, the rapid burnup of gadolinium can be mitigated by concentrating it in a small number of pins in each fuel assembly [25]. Using a high concentration of gadolinium in a small number of fuel elements, rather than evenly distributing it, results in self-shielding of the Gd which can extend its burnup [27]. Additional calculations were therefore performed to examine the burnup characteristics of fuel with BNA incorporated in only 8 out of the 32 pins in each of the inner and outer rings, depicted schematically in Figure 9.

Results of the lattice cell calculations showing the infinite lattice neutron multiplication factor as a function of burnup (at the midpoint of the channel) for three concentrations of gadolinia, are shown in Figure 10. Based on these results, the concentration of 4% gadolinia in 16 fuel pins provided adequate reactivity suppression with little decrease in lattice exit burnup. The BNA will be located in eight evenly-spaced pins in the outer ring and eight evenly-spaced pins in the inner ring of the fuel assembly. When compared to the case with no BNA in the fuel, the burn-out period for the gadolinia appears to be slightly more than 40 MWd/kg, i.e., when the lattice reactivity returns to that of fuel with no BNA.

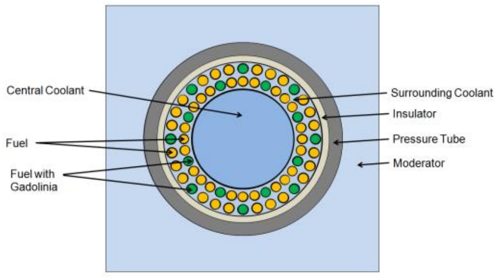

Figure 9: Cross-Sectional View of the 64-Element Canadian SCWR Fuel Assembly Concept Incorporating Burnable Neutron Absorber, Channel, and Lattice Cell

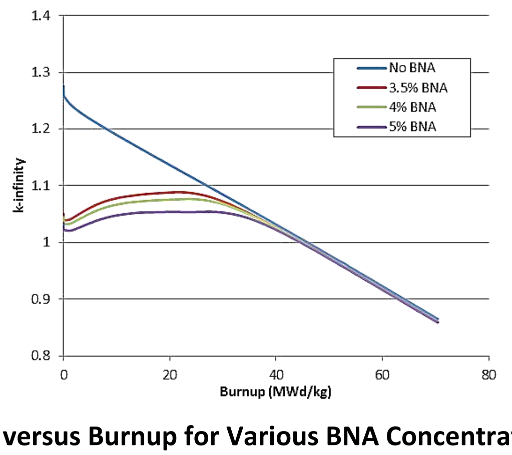

Figure 10: Lattice K-Infinity versus Burnup for Various BNA Concentrations in the Canadian SCWR

Without reactivity suppression, there is a large, > 100 mk, excess reactivity at the beginning of each fuel cycle. Despite the large initial excess reactivity, a judicious choice of refuelling scheme limits the maximum channel peaking factor to 1.3 at the beginning of cycle. The integral core channel power distribution is shown in Figure 11. The axial power distributions at the beginning of cycle (BOC) and end of cycle (EOC) are shown in Figure 12.

	1	2	3	4	5	6	7	8	9	10	
A							0.85	0.98	0.79	1.07	BOC
							0.94	1.06	0.84	1.12	EOC
B					0.95	0.91	1.03	0.82	1.07	0.85	
					1.07	1.00	1.12	0.87	1.12	0.88	
C			0.73	0.65	0.71	1.07	0.87	1.12	0.88	1.19	
			0.88	0.76	0.80	1.14	0.90	1.15	0.90	1.19	
D			0.65	0.92	0.92	1.03	1.03	0.92	0.99	1.01	
			0.76	1.05	1.00	1.06	1.03	0.91	0.99	1.02	
E		0.95	0.71	0.92	0.92	1.29	1.13	1.24	1.02	0.87	
		1.07	0.80	1.00	0.92	1.22	1.06	1.17	0.99	0.85	
F		0.91	1.07	1.03	1.29	1.03	1.31	0.94	0.90	1.21	
		1.00	1.14	1.06	1.22	0.94	1.19	0.87	0.84	1.13	
G	0.85	1.03	0.87	1.03	1.13	1.31	0.97	1.07	1.24	0.94	
	0.94	1.12	0.90	1.03	1.06	1.19	0.88	0.99	1.14	0.87	
H	0.98	0.82	1.12	0.92	1.24	0.94	1.07	1.13	1.09	1.07	
	1.06	0.87	1.15	0.91	1.17	0.87	0.99	1.05	1.02	1.00	
J	0.79	1.07	0.88	0.99	1.02	0.90	1.24	1.09	0.95	1.07	
	0.84	1.12	0.90	0.99	0.99	0.84	1.14	1.02	0.89	1.01	
K	1.07	0.85	1.19	1.01	0.87	1.21	0.94	1.07	1.07	1.10	
	1.12	0.88	1.19	1.02	0.85	1.13	0.87	1.00	1.01	1.03	

Figure 11: Quarter Core Channel Map of Normalized Channel Power Distribution for BOC and EOC of the Device Free Core without BNA, for the Canadian SCWR

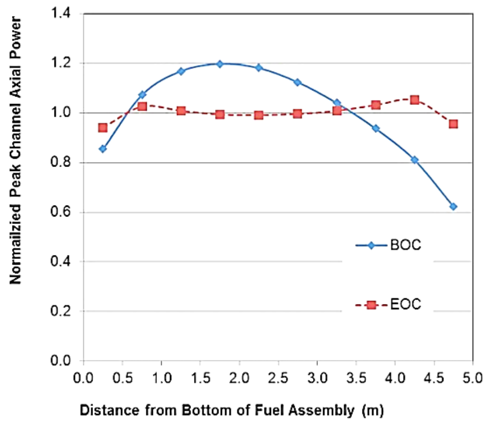

Figure 12: Channel Axial Power Profile for BOC and EOC of the Device Free Core without BNA, for the Canadian SCWR

Stainless steel absorber rods, passing horizontally through the core, are used for further reactivity suppression and power shaping. These rods are similar to those used in a conventional HWR and are composed of SS-316 stainless steel. The rods pass through the core on the north and south faces, and are "half-length" (i.e., at maximum insertion reach the centre of the core). With these rods, it is possible to obtain a flatter channel power distribution than in the absence of BNA, while achieving adequate reactivity suppression during operation.

The positions of the adjuster rods (Figure 13) correspond to the meshes used in the Reactor Fuelling Simulation Program (RFSP) models, rather than the actual rod geometries. The front view of the reactor is shown in Figure 13B, illustrating the five banks of seven horizontal rods. On the opposite side of the reactor are another five banks of seven rods, making a total of 14 rods that can be moved independently. The top view, showing all rods inserted, is in Figure 13A.

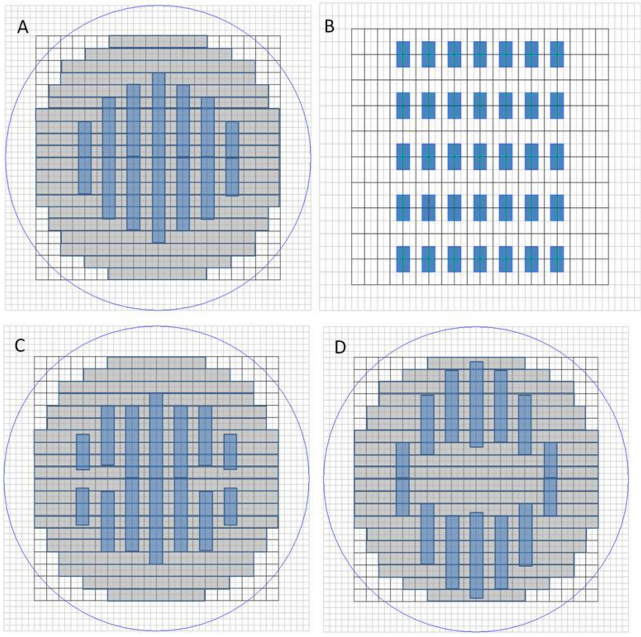

Figure 13: Cross-Sectional View of the Full Core Channel Layout, Showing Positions of Inserted Horizontal Reactivity Devices. A) Top View Adjuster Rod Reference Positions. B) Front View Showing Five Banks of Rods. C) Middle Bank of Rods at BOC Position. D) Middle Bank of Rods at EOC Positions

As with adjuster rods in a conventional HWR reactor, these rods sit in the core during normal operation and are moved as necessary to provide flux shaping. Figure 13C and D show middle bank rod positions at the beginning and end of cycle, respectively.

The Canadian SCWR concept has two independent emergency shutdown systems, following the practice presently employed in conventional PT-HWRs. The first shutdown system, SDS1, is a set of neutron-absorbing rods, which are inserted in the core in response to a reactor trip. The rods will be modelled after those currently used in PT-HWRs. Two options for rod orientation are currently being considered: horizontal and diagonal. A horizontal rod configuration has the advantage of simplicity in concept development and analysis; interference with the adjuster rods and other in-core devices is minimized and reactor physics modeling of the SDS1 system can be based on conventional techniques. A diagonal rod configuration has some appeal since there is the possibility to credit gravity for rod insertion, given a loss of the rod drive mechanism.

The second emergency shutdown system, SDS2, is similar to that employed in conventional HWRs and based on the injection of soluble neutron absorber (i.e., neutron poison) into the moderator [28]. Accurate modeling of the reactivity effect of poison injection, especially at intermediate injection times, requires full 3-dimensional modeling that captures the shape of the injection jets. Monte Carlo N-Particle Transport Code (MCNP) 3-D stochastic neutron transport modeling, based on a combination of prior computational fluid dynamics and empirical modeling, was used to assess the use of SDS2 in the Canadian SCWR concept. A schematic view of a CFD and corresponding MCNP model of a poison jet in the core are shown in Figure 14.

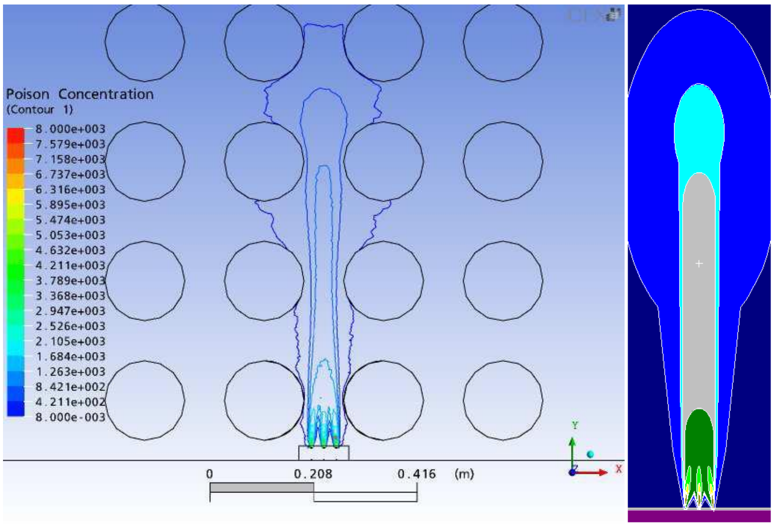

Figure 14: Schematic View of CFD & MCNP Poison Jet Models

The impacts of the spacing between jets and choice of jet direction were examined, and it was shown that the SDS2 option is feasible for use in the Canadian SCWR concept. The calculated change in reactivity was observed to drop by more than 100 mk in less than two seconds (the drop increases by ~15 mk for each channel pitch the injectors are spaced apart; peak at ~135 mk). The reactivity reduction achieved compares favorably with the current poison injection systems in existing HWRs.

A sudden control rod movement in a Canadian SCWR will cause a strong perturbation in the power distribution, which will be flattened later by the feedback effect. The hot channels were used to evaluate the compliance with the acceptance criteria for this transient. Cladding and fuel centreline temperatures were examined in the hot channels of three assemblies. The maximum cladding temperature (860ºC) slightly exceeds the criterion for buckling the cladding, but does not exceed the limit of 1200°C when severe oxidation of the cladding is expected. The maximum centerline fuel temperature (2600ºC) remained below the melting point of 2800°C.

In the HPLWR design, a two-pass core is used, heating the coolant in fuel assemblies in the outer core region with a downward flow and an upward flow in the inner core region. The flow configuration is shown in Figure 15. In the reactor design, the core barrel, control rod drives and control rod guide tubes are taken from PWR design, except the reactor pressure vessel accounts for the 25 MPa pressure by using a thicker wall. Details of the pressure vessel design and its structural analysis have been published by Fischer et al. [29]. A higher core outlet temperature is achieved with a three pass core, shown on the right hand side of Figure 15. In this configuration, the feed water enters the core from below, flowing upwards (shown by a blue and red thick arrow) in the central fuel assemblies of the core and the heated coolant is mixed in a steam plenum above the core. A second heatup step is provided in a downward flow (shown by a red and yellow thin arrow) through a set of fuel assemblies surrounding the central fuel assemblies. The flow is again mixed in an annular mixing plenum underneath the core, and heated to the core outlet temperature with an upward flow (shown by a yellow and white thin arrow) at the fuel assemblies in the periphery of the core.

Flow stability problems caused by the large density change of the boiling coolant in the core are well known from BWRs. Stability analyses of two and three pass cores show that they are effectively avoided by the installation of orifices at the inlet of fuel assemblies, customized for a hot fuel assembly. The stability guidelines of BWRs were extended for the fuel assemblies of HPLWR heat-up components (evaporator, superheater I and superheater II). The analyses completed show that hot fuel assemblies of the superheater to satisfy stability criterion without any orifices [6]. For the fuel assemblies of the evaporator stage must be equipped with inlet orifices.

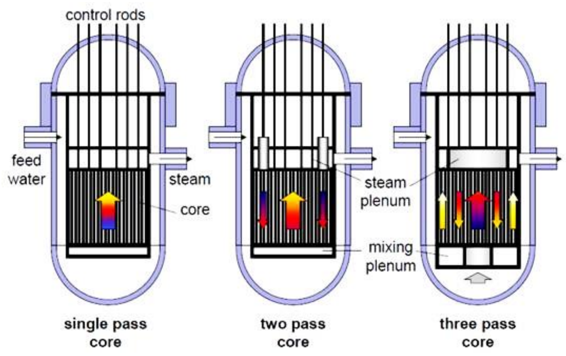

Figure 15: Different Core Flow Configurations Used in Heating the Coolant in the HPLWR

Five different types of fuel assembly clusters are used in the HPLWR design during first full loading, with fresh fuel having up to 7% ^{235}U enrichment and 2% Gd poisoning. Figure 16 shows one fourth of the core and the five fuel cluster types used in HPLWR, numbered 1 to 5 in white colour. Surrounding the white-numbered rods are eight more rods forming a nine-rod fuel cluster. The four corner rods in each fuel cluster use 1% lower enrichment than the other fuel rods in the cluster. As indicated in red in Figure 16, there are seven fuel clusters with control rods inserted for burn-up compensation. The corner rods of each cluster are not equipped with control rods. The black solid lines separate superheater 1 from evaporator and superheater 2 assemblies. Dashed white lines indicate the assembly clusters. The radial form factor, i.e. the individual assembly power, normalized with the average assembly power of each heat-up step, has a uniform distribution up to around 1.25 with the exception of 2 assemblies in superheater 2 which exceed a form factor of 1.3. The local assembly power is reduced at control rod positions, causing a higher power in assembly clusters without control rods, namely by 5.6% in evaporator clusters and by 4% in clusters of superheater 1.

From calculations shown in Figure 17, the power is the highest in the evaporator, where more than 1400 MW of the total thermal power of 2300 MW is supplied at the beginning of the cycle, whereas the superheater 2 supplied only around 100 MW. During burn-up, the evaporator power decreased as fuel is consumed at a faster rate in that region; only 1300 MW is produced in the evaporator at the end of the cycle, whereas the power in the superheater 2 increased to 150 MW. The Fuel Assemblies of the HPLWR are 6.175 m long and are segmented into 39 layers for analysis.

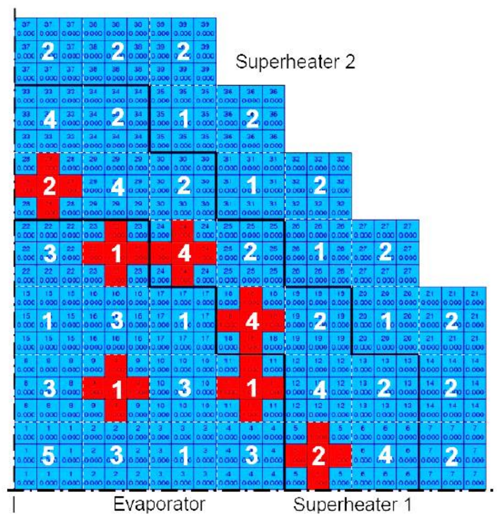

Figure 16: Control Rod Positions for Reactivity Compensation of Fresh Fuel in the HPLWR Design

As shown in Figure 18, the neutron multiplication factor k_{eff} with fully withdrawn control rods reduced to 1 within about 1 year at full power. To compensate local power peaks, at least partially, the hotter assemblies are cooled with a higher optimized coolant mass flow.

The hottest fuel temperature in the HPLWR is expected to occur in the evaporator, where the coolant temperature is relatively low, but the assembly power is the highest. The hot spot of the fuel centerline temperature and the maximum linear heat rate were predicted to occur in superheater 1. The fuel centreline temperature of 2117°C and 1883°C were predicted for beginning and end of fuel cycle of an equilibrium core, respectively, corresponding to a maximum linear core power of 365 W/cm and 305 W/cm.

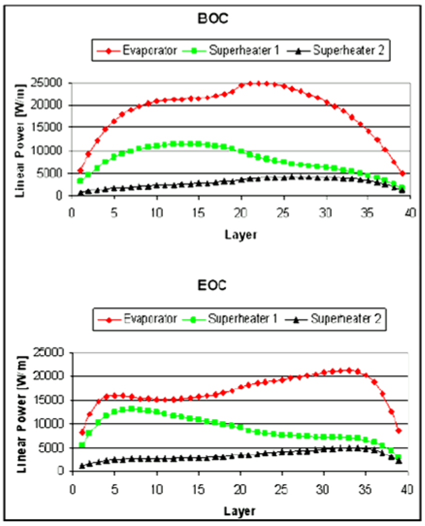

Figure 17: The Linear Power Profile Along the Axial Direction of Fuel Assembly Segmented into 39 Layers

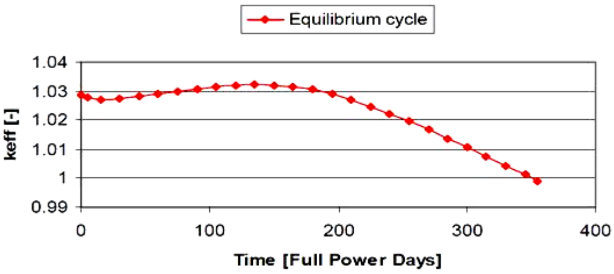

Figure 18: The Effective Neutron Multiplication Factor during an Equilibrium Burn-up Cycle with Control Rods Withdrawn

The consequences of a control rod ejection was analysed for HPLWR. The accident analysed was a mechanical failure of a control rod mechanism housing that leads to the ejection of a control rod up to its uppermost position, driven by reactor vessel pressure. The consequence is a fast and large positive reactivity insertion and the excursion of core power with a large localized relative power increase. The objective of the analysis was to determine whether damage to either the fuel or the coolant pressure boundary occurs due to the power excursion. The analysis was performed using two set of coupled codes; TRAB-3D with SMABRE, and ATHLET with KIKO. The ejected control rod was in superheater 1 close to the evaporator. In the slow ejection case (1 s) the results are practically identical for the two sets of coupled codes. In the case of fast ejection (0.1 s), SMABRE/TRAB-3D predicted a larger insertion of reactivity, with a very short duration peak. Consistent with the reactivity, the peak in the power calculated with SMABRE/TRAB-3D was higher than the one calculated with ATHLET-KIKO3D. In general, however, the results were in excellent agreement with each other and provide sufficient confidence that the results obtained for other transients are representative of the behavior of the HPLWR.

6.1.2 Risk of re-criticality

The defense-in-depth approach to the development of the Canadian SCWR includes avoidance of re-criticality after postulated accident scenarios, regardless of the configuration of the core or perturbations to it. Three broad scenarios are considered in the evaluation of re-criticality: no core damage; partial core damage; and complete core disassembly.

In the case of no core damage, only one condition results in positive reactivity insertion, that is, voiding of the coolant surrounding the fuel, while the coolant in the central flow tube remains unperturbed. In practice, such a scenario is only possible for a brief period during a transient and results in a small positive reactivity insertion, ~ 1-2 mk, which is within the nuclear data uncertainty evaluated for SCWR coolant void reactivity. If the emergency liquid injection system was activated during the postulated accident, over 100 mk of negative reactivity insertion would suppress the effect of partial voiding.

For partial core damage, it is assumed that one or more pressure tubes have been damaged, due to an initiating small-break loss-coolant event resulting in coolant flow into the moderator. In this case, the higher neutron absorption cross section of light water would result in a negative reactivity insertion.

In the case of complete core disassembly, following a station blackout event, it is assumed that the fuel melts along with core structural materials to form corium. In the situation where no cooling water is provided, the corium would be subject to a fast neutron spectrum (no moderation). The presence of burnable neutron absorbers, non-volatile fission products, stainless steel cladding material, and fertile isotopes (e.g. Th-232) will all suppress criticality in a fast neutron spectrum; even if the material were compressed into a sphere, that sphere would not have sufficient neutron multiplication to become critical. Corium submerged in light water could be viewed (in a worst case scenario) as a corium sphere with light water reflector (containing no neutron absorber). As in the fast spectrum case, the presence of absorbing materials would suppress neutron multiplication, which is more significant than the neutron multiplication gained through moderation in the cooling water. The negative Fuel Temperature Coefficient provides negative feedback under postulated high temperature accident scenarios, thus supporting the passive safety concept of the SCWR. As such, the Canadian SCWR is not expected to return to critical under any post-accident condition, without significant human interference.

6.2 Decay heat removal

The safety approach adopted for the SCWR concept follows those of advanced reactors in that multiple levels of independent and diverse safety systems provide defence-in-depth, and passive safety systems are adopted for increased reliability. One of the major development goals of the SCWR concept is to enhance safety, such that the risk of core damage and release of radioactive materials to the environment is significantly reduced. The unique features of the pressure-tube based Canadian SCWR concept allow for an optimum balance of passive safety features in the moderator systems for emergency heat removal (such as a prolonged station blackout event), and a combination of active and passive safety systems in the main cooling system. The primary system components are selected to provide multiple and redundant decay heat removal paths; these defence-in-depth concepts should provide improvements in plant risk over existing reactors. However, there is a transformative improvement in reducing core damage risk by including a further passive decay heat removal pathway for emergencies. This capability is made possible through a combination of a natural-circulation-driven moderator cooling system, the fuel assembly concept, the fuel channel concept, and direct radiation heat transfer from the fuel to the insulator liner.

The safety concepts adopted for the Canadian SCWR concept are described in reference [30], with detailed descriptions of the safety systems in references [31] and [32]. The decay heat removal capabilities of the SCWR were assessed through an analysis of the station blackout scenario [33] representing the early days of the accident. A typical response from the analysis is shown in Figure 19. Isolation condenser heat exchangers, immersed in the reserve water pool, are used to dump core heat to the reserve water pool, in case of loss of grid power, via natural circulation of the core coolant. The isolation condenser system is shown in Figure 20. The analyses indicate that natural circulation cooling through isolation condensers is feasible [34]. To avoid thermal shock on the condensers suddenly being filled with hot primary coolant while they are immersed in reserve water pool at ambient temperatures, the strategy is to first vent the coolant to a suppression tank for a short time, to bring the pressure and the temperature down to below the critical point, and then to open the valves to the isolation condensers to activate the passive cooling loop. During this period, the decay heat transferred from the isolation condensers to the reserve water pool will increase the pool temperature to saturation (15 h) and induce boiling. During the following 3 days, the water level in the reserve water pool will decrease through evaporation, but the isolation condensers and the air-cooled heat exchanger system placed on the floor of the water pool will still be covered by water, so that the heat transfer path from the reserve water pool to the environment will still be maintained. The decay power calculated for the Canadian SCWR is shown in Figure 19 as a percentage of full power, 2540 MWt. The cumulative water evaporation (at atmospheric pressure) needed to remove this decay heat is shown on a secondary y-axis in the same figure. About 1800 m^3 of water evaporates from the reserve water pool, as decay heat is removed over the first 3 days. After 3 days, the decay heat reduces to less than 0.5% of the full power (12.5 MWt) and air-cooled heat exchangers alone become sufficient to remove decay heat with no further boiling in the reserve water pool. The location of the air heat exchangers are shown in Figure 21. The reserve water pool is conservatively sized to contain 5000 m^3 of water.

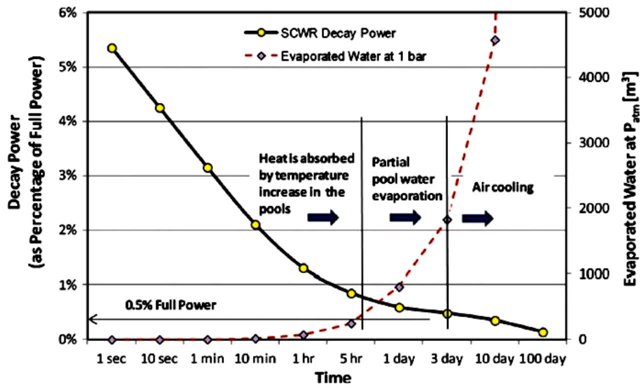

Figure 19: Calculated Decay Power and Pool Water Requirement to Boil Off During Accidents

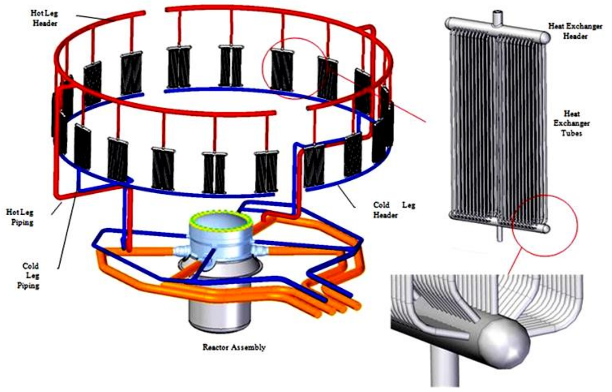

Figure 20: Isolation Condenser System

Another decay heat removal path designed into the Canadian SCWR is the no-core-melt concept. This was added to meet the enhanced safety requirements outlined by the GIF [2]. The fuel channel design for the Canadian SCWR constrains the peak temperature of the fuel sheath/cladding below its melting temperature, following a loss of coolant accident (LOCA) with loss of emergency core cooling (LOECC). This constraint, termed no-core-melt, aims to reduce or eliminate fission product release under such postulated accident scenarios, thereby reducing the negative consequences of such accidents. The insulated pressure tube concept, termed High Efficiency Channel (HEC), was identified as a viable concept for mitigating the consequences of a LOCA/LOECC [35].

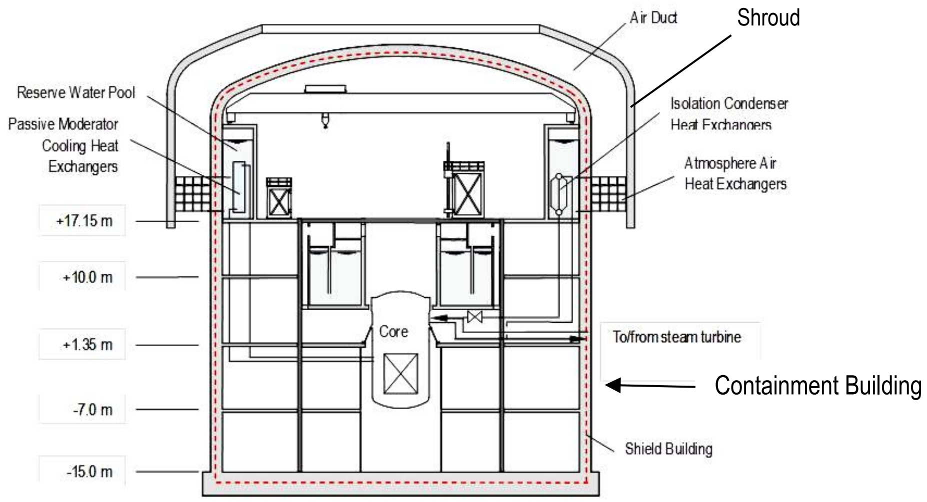

Figure 21: Cross-Sectional View of the Shield and the Containment Buildings in the Canadian SCWR Concept

In the HEC concept, the pressure tube is in direct contact with the moderator, and acts as a pressure boundary between the primary coolant and the moderator as shown in Figure 22. The pressure tube is thermally isolated from the primary coolant by an encapsulated insulator to ensure that the pressure tube outer surface temperature is maintained below the saturation temperature of the moderator fluid during normal operation. The insulator is clad with an inner and outer liner tube; the inner liner tube is in direct contact with the primary coolant. To minimize the temperature of the fuel cladding during a postulated LOCA/LOECC scenario, sufficient decay heat must be transferred from the fuel to the moderator. However, to enhance the thermal efficiency of the power cycle during normal operation, radial heat loss from the fuel/coolant to the moderator should be minimized. Thus, the HEC geometry is optimized to reduce the radial heat loss from the fuel channel during normal operation and to increase the radial heat loss during LOCA/LOECC accident scenarios.

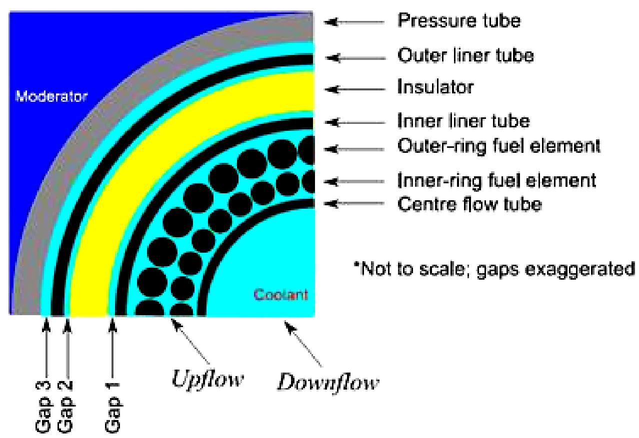

Figure 22: Cross-Section of the Fuel Channel for the Canadian SCWR

Using the CATHENA thermalhydraulics code, simulations and experiments were conducted to ensure the sheath temperature remains below its melting temperature under long term cooling following a LOCA/LOECC. Assuming the fuel channel voids instantaneously after a LOCA/LOECC event, an analysis was conducted to establish the peak component temperatures during the simulation. The peak fuel sheath temperature was observed to occur approximately two minutes after LOCA/LOECC. The calculated temperatures along the radius of the fuel channel are shown in Figure 23. The most significant heat transfer contribution from fuel to pressure tube occurred via thermal radiation between intermediate components along the radius. As can be seen in the figure, the no-core-melt concept is feasible in the Canadian SCWR, even under the worst-case assumptions made in the analysis. Further studies continue to improve the design and the materials for the insulator and the liner tubes.

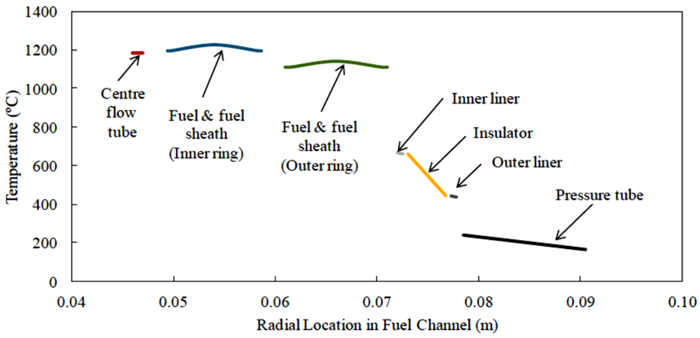

Figure 23: Temperature Distribution of Fuel and Fuel Channel Following a LOCA/LOECC Event for 0.75 mm Gap between Pressure Tube and Outer Liner (the fuel sheath temperature represents the maximum reached in the transient 2 minutes after the initiation of the event)

6.2.1 Thermal inertia and grace period

Thermal inertia represents the ability of a material to conduct and store heat. Thermal inertia is reciprocal of thermal response. It is defined as the rate of temperature rise in a reactor operating at its rated power when no heat is removed by primary or other cooling systems. High thermal inertia gives slow thermal response and allows the reactor to tolerate loss of cooling for a long time. If a reactor has an appropriately large thermal inertia, it will take days before decay heat raises the reactor temperature sufficiently to cause fuel failures without cooling. This gives: (1) adequate time for operator action, (2) reduces the requirements for the decay heat removal system, and (3) provides time for short lived radionuclides to decay away. As shown in Figure 19, for approximately 1 h very little water is required for evaporation in the Canadian SCWR. Most of the decay heat in this period is absorbed by bringing the reserve water pool temperature to saturation. Subsequently, about 1000 m^3 of water is evaporated within the first day, making the accident progression slow and allowing adequate time for operator intervention.

Dragunov [36] compared the thermophysical properties of coolants proposed for the Gen IV systems to differentiate their relative thermalhydraulic advantages. He proposed that thermalhydraulically, the best coolant would remove the largest amount of heat per unit pumping power required to force the fluid against the hydraulic resistance within the core. In order to perform this comparison, Dragunov calculated the ratio of i) total heat removed by the coolant from the core to ii) the power required to circulate the coolant to reach an outlet temperature. The total heat removed ($\dot{Q}$) from the coolant was calculated by multiplying the mass flow rate of coolant ($\dot{m}$), specific heat of the coolant (C_p), and the coolant temperature difference between the inlet and outlet of the core (ΔT). The pumping power required ($\dot{W}$) was calculated by multiplying the mass flow rate of the coolant with the hydraulic resistance of the thermalhydraulic loop (ΔP) and dividing it by the density of the coolant (ρ). The ratio, $\frac{\dot{Q}}{\dot{W}} = \frac{\dot{m}\,\rho\,C_p\Delta T}{\dot{m}\,\Delta P}$. When rearranged, the factor, $a = \frac{\dot{Q}\,\Delta P}{\dot{W}\,\Delta T} = \rho C_p$. For a fixed hydraulic resistance and allowing the exit temperature to increase as required, the coolants can be compared based on the quantity "a". Dragunov calculated this factor for SCW, helium (at 7 and 9 MPa pressure), CO_2, sodium, lead, and the lead-bismuth mixture for the temperature range of 350 to 650°C. The behaviour of specific heat and the density for the calculated coolants are shown in Figure 24. The calculated results for 350ºC are shown in Table 2.

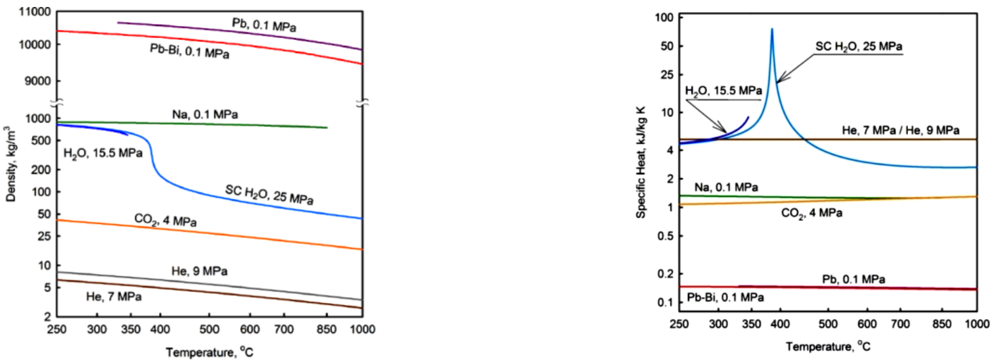

Figure 24: The Behaviour of Density and Specific Heat for Various Coolants within the Temperature Range between 250 and 1000°C [36]

As can be seen in the last column of Table 2, SCW greatly outperforms other coolants in the 350ºC temperature range. There are several other factors such as corrosive properties, cost of pressure boundary, size and cost of the machinery for the power conversion side of the plant, and the net-plant efficiency that may alter the coolant choice of a reactor design. The analysis made above shows that SCW is an efficient coolant for decay heat removal at 350°C.

Table 2: A Comparison of Required Coolant Power for Select Number of Coolants [36]

Coolant	Specific Heat (kJ/(kg.K)	Density (kg/m^3)	a (kPa/K)	$a_{scw}/a_{coolant}$
SCW	6.713	626	4,202.0	1
He, 7 MPa	5.189	4.6	23.8	176
He, 9 MPa	5.187	5.3	27.5	153
CO_2	1.2	34.2	41.0	102.4
Na	1.4	900	1,260.0	3.3
Pb	0.16	10,500	1,680.0	2.5
Pb-Bi	0.16	10,200	1,632.0	2.6

The thermal inertia (*I*) is calculated as the square root of the product of specific heat, thermal conductivity and density of a material. The thermal conductivity of various coolants is provided in Figure 25. The coolant thermal conductivity values at 350ºC were used in calculating the thermal inertia as shown in Table 3. From a thermal inertia perspective, sodium is six times higher than SCW.

Another 5,000 m^3 reserve water pool (Figure 21) serves as a buffer between the various passive safety systems and the ultimate heat sink. This large mass of water allows for temporary absorption of heat, which can be subsequently removed by the atmospheric air heat exchangers or by evaporation. This volume of water will also provide thermal inertia to the reactor. The primary function of the reserve water pool is the operation of Isolation Condensers (IC), which is designed to remove sensible and core decay heat from the reactor passively, preventing reactor overpressure and to serve as a long-term cooling system under station blackout conditions. The isolation condenser heat exchangers connect with the reactor coolant piping, and remove heat from the reactor by depositing it into the reserve water pool (Figure 21). The isolation condenser system is divided into two independent trains, with each train consisting of a piping loop running from the reactor outlet, to heat exchangers located in the reserve water pool, and returning to the reactor inlet (see Figure 20). The system is pressurized and on hot standby under normal reactor operations. A connection valve is located on the system's low point near the reactor inlet, and is closed under normal reactor operations. During a station blackout sequence, the reactor is depressurized and cooled by closing the main steam and feed-water isolation valves, followed by opening the IC connection valve. IC flow is driven by the density difference between the IC hot leg and cold leg fluid to initiate and maintain a gravity-driven circulation (Figure 20).

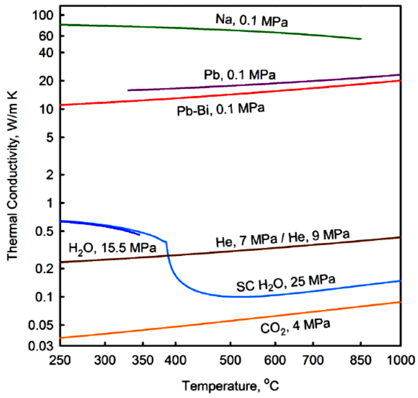

Figure 25: The Characteristics of Thermal Conductivity for Various Coolants between 250 and 1000°C [36]

Table 3: Thermal Inertia of Coolants Proposed for Gen IV Reactor Systems at 350°C

Coolant	a (kPa/K) from Table 2	Thermal Conductivity, k (W/(m.K))	Thermal Inertia, I = $\sqrt{ak}$ (N/(m.K.s^{½}))	I/I_{SCW}
SCW	4,202.0	0.5	1450	1
He, 7 MPa	23.8	0.3	84.7	0.06
He, 9 MPa	27.5	0.3	90.8	0.06
CO_2	41.0	0.045	43	0.03
Na	1,260.0	60	8695	6
Pb	1,680.0	16	5185	3.6
Pb-Bi	1,632.0	11.5	4332	3

6.2.2 Diversification, Active and Passive Systems

During the course of the developmental work of the Canadian SCWR concept, only the passive safety systems were identified and incorporated. The active safety systems serve the same function and operate under similar conditions as those found in the current Generation II and III reactors, and thus current technology can be readily adapted for this purpose. For this reason, the active systems are not discussed in this paper unless they are new and / or perform a new function.

The first component demonstrating diversification is the containment pool shown in Figure 26 that consists of an annular shaped tank located in the containment building above the reactor. The liquid level during normal operating conditions is 13.0 m above the top of the fuel, with a pool depth of 6.85 m. The pool is divided in two sections, to reflect the bilateral symmetry of the reactor and safety systems, each half functioning independently of the other. In order to assure long-term decay heat removal, in the event of a piping breach within the containment building steam tunnel (Figure 26), the volume of the containment pool exceeds that of the steam tunnel. Due to the seal between the reactor and steam tunnel floor, coolant will accumulate within the steam tunnel. A sufficient level of water will remain in the containment pool to cover both the suppression nozzles and the gravity driven core flooding system inlet pipe. This feature eliminates the need for an active pumping system, and will not require sump strainers.

Located above the liquid level within the pool is the second component of diversification consisting the containment steam condenser gallery, which houses containment steam condenser heat exchangers and passive autocatalytic recombiner (PAR) units (Figure 26). Physically, the condenser gallery is an annular shaped, enclosed area, with a series of openings located on the outer wall. This outer wall forms a separation between the steam tunnel and condenser gallery. Located within these openings are the containment steam condensers, placed to allow condensed steam to drain directly to the condenser gallery. The condenser gallery floor has a series of drains with suppression nozzles, discharging into the containment pool below the liquid level. This layout permits the containment steam condensers and containment pool to act in unison to condense steam in the steam tunnel during a high-steam-flow-LOCA period, allowing steam to flow past the condensers and be injected and suppressed within the containment pool via the drains. A low-steam-flow period will result in the direct condensation of the steam by the heat exchangers, with the condensate draining into the containment pool. The volume above the liquid level of the containment pool can be considered as a wetwell. In a period of high-steam-flow from the steam tunnel to containment pool, air and gases may be entrained, and deposited in the wetwell above the surface of the containment pool. In order to prevent the pressure in this area from rising excessively, a series of rupture panels are located above the containment pool water line, separating the drywell space from the wetwell. These allow gases and entrained air to escape to the larger drywell space, should the wetwell volume become insufficient.

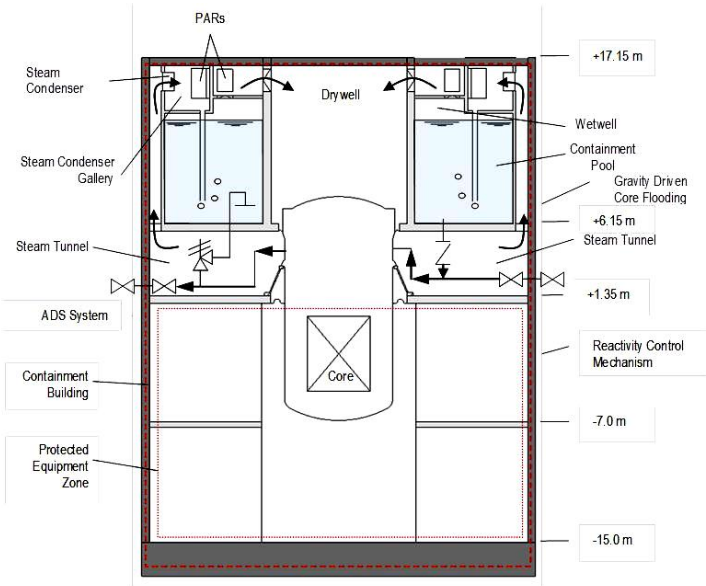

Figure 26: Cross-Sectional View of the Canadian SCWR Containment Building

The third component of diversification is the reserve water pool (Figure 21) to which the secondary side of the containment steam heat exchangers (Figure 26) are connected, with circulation established via gravity-driven flow. With this, heat from a LOCA event will be deposited into the reserve water pool via the containment steam condensers. The primary function of the reserve water pool is to serve as a buffer between the various passive safety systems and the ultimate heat sink (large water pool). The large mass of water available can temporarily absorb heat, which can be subsequently removed by the atmospheric air heat exchangers or by evaporation. The pool is located in the upper section of the shield building (Figure 21), and occupies an annular space against the building outer wall and contains a gross water volume of 5,000 m^3. This is divided into two sections, each section housing one train of the isolation condenser and passive moderator cooling systems. All heat exchange areas of the isolation condenser and passive moderator heat exchangers are located in the lower half of the pool, allowing approximately 2,100 m^3 of pool water to be lost to evaporation, while still functioning as a heat sink for the isolation condensers and passive moderator cooling system. The pool enclosure is equipped with a filtered vent to the atmosphere in order to permit the release of water vapour. Pool levels can be remotely maintained by means of a fill line connected to an external emergency supply such as lake water or trucked water.

The fourth component of diversification is the isolation condensers (Figure 20) which remove sensible and core decay heat from the reactor passively, preventing reactor overpressure and serving as a long-term cooling system under station blackout conditions. The functional details of the IC are described in Section 6.2.1.

The fifth component of diversification is the ADS that consists of several valves through which the reactor can be rapidly depressurized. In addition, the ADS system provides overpressure protection to the reactor and outlet piping. The valve banks are located in the containment building steam tunnel, with the discharge flow suppressed into the containment pool (Figure 26).

The sixth component of diversification is the gravity-driven core flooding system (Figure 26) consisting of a pipe connecting the containment pool to the reactor cold leg coolant piping. A check valve permits the reactor to operate at its operating pressure, yet allow water to flow into the reactor from the containment pool under accident conditions.

The seventh component of diversification is the atmospheric air heat exchangers (Figure 21) to reject heat from the reserve water pool to the atmosphere. Although not considered a safety system, the heat exchangers serve to extend the period of time during which the reserve water pool can function as a heat sink before intervention, during a high-core-decay heat period. At lower core decay heat, the atmospheric air heat exchangers can reject the entire heat load, extending indefinitely to the point of intervention. The atmospheric air heat exchangers consist of a series of plate-type heat exchangers located on the periphery of the shield building (Figure 21) for a total heat exchange area of 58,000 m^2. These heat exchangers are enclosed in a shroud having an inner diameter of 57 m, which forms a chimney to further increase gravity-driven air flow. In order to minimize the number of penetrations into the shield building, the heat exchangers are grouped and connected to a common hot leg and cold leg headers. Valves are located on both the hot leg and the cold leg headers and are closed under normal reactor operating conditions to prevent freezing in cold climates. Under accident conditions, with the valves opened, water is drawn from the upper surface of the pool, cooled in the heat exchanger, and returned to the bottom of the pool by a gravity-driven convection current. Similarly, cooler air is drawn through the heat exchangers from the bottom of the shroud, with the heated air escaping at the top of the shroud.

The eighth and final component of diversification is the passive moderator cooling system (PCMS), which serves as an additional barrier to core damage (Figure 27). In an accident scenario, decay heat generated within the fuel channel flows through the channel insulation and pressure tube, and is deposited into the moderator. The passive moderator cooling system uses a flashing-driven natural circulation loop to remove heat from the moderator, and deposit this into the reserve water pool. The passive moderator cooling system is divided into two independent trains, with each train consisting of a piping loop running from the reactor calandria to heat exchangers located in the reserve water pool, and returning to the calandria (Figure 27). The system is totally passive, and functions during normal reactor operation. A head tank, located above the heat exchangers, maintains a constant pressure within the system, with the liquid level 25.5 m above the top of the fuel elements.

In the HPLWR design, diversification is achieved through a number of safety systems. The first diversification is that HPLWR removes residual heat by forced convection inside the reactor, which is driven by a natural convection loop outside, but the requirement for the safety system, in general, is to ensure sufficient coolant mass flow rate. The second diversification is in the primary shutdown system, which uses control rods, and by a boron injection system used as a second, independent shut down system. The third diversification is the containment isolation by active and passive containment isolation valves in each line penetrating the containment, to close the containment in case of an accident. The fourth diversification is steam pressure limitation by pressure relief valves. The fifth diversification is the automatic depressurization of the steam lines, into a pool inside the containment through spargers, to close the coolant loop inside the containment in the case of containment isolation. The sixth diversification is a coolant injection system to refill the pressure vessel after intentional or accidental coolant release into the containment. The seventh diversification is a pressure suppression pool, to limit the pressure inside the containment in the case of steam release inside the containment. The eighth diversification is a residual heat removal system for long term cooling of the containment.

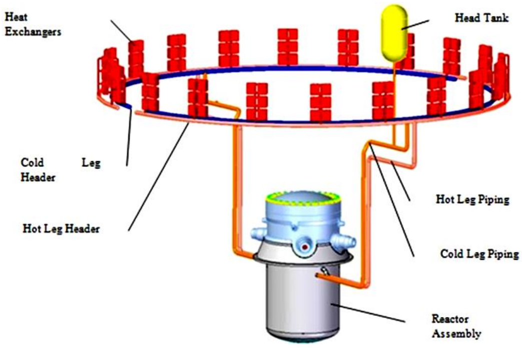

Figure 27: Passive Moderator Cooling System

6.3 Confinement of Radioactive Materials

The containment building for the Canadian SCWR is a double-wall containment with integrated passive safety features within the containment. The containment system includes multiple and diverse, passive and active, engineered safety features to meet GIF safety and reliability goals. The design has an engineered passive cooling capability for up to 72 hours to ensure the ultimate heat sink is available without operator intervention and external power supply. As a Generation IV reactor, the need for offsite emergency response has been eliminated and indefinite passive cooling of the containment to the atmosphere as the ultimate heat sink has been provided. The decay power of the reactor reduces to less than 1% within a day after shutdown, and passive containment cooling to the ultimate heat sink is feasible.

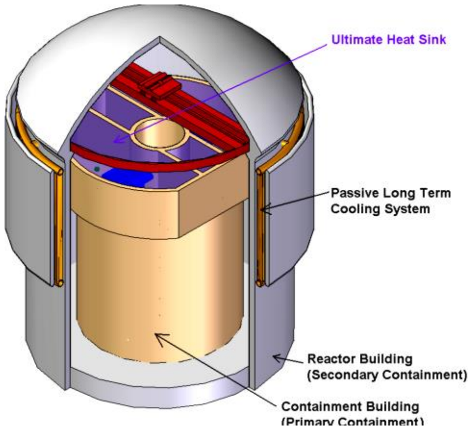

Figure 28: The Reactor Building

The pre-conceptual design of the Canadian SCWR containment design includes a double-wall containment where the primary containment is enveloped by a larger reactor building (Figure 28). The reactor building will house the containment building, the ultimate heat sink (large water pool) as well as the safety-related mechanical components and the required support and protection systems. Figure 28 shows an image of the recommended containment concept for the Canadian SCWR. The containment building (primary containment) will envelop the components of the reactor pressure boundary, or those connected to coolant pressure boundary, that cannot be isolated in the event of an accident.

The containment is designed to reduce the potential for contamination release to the environment by the creation of a steam tunnel and high pressure turbine containment, since super critical water flow to the high pressure turbine may carry contamination. The reactor containment, the steam tunnel and the high pressure turbine containment are shown in Figure 29.

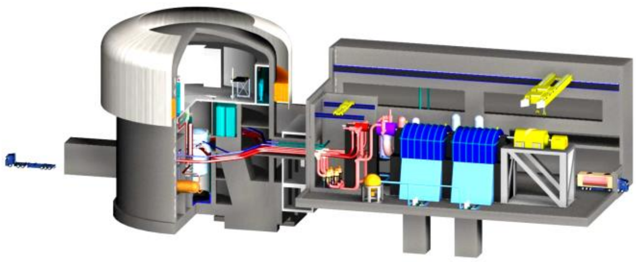

Figure 29: Reactor Buildings and the Turbine Building

The Canadian SCWR containment building will also envelop the safety systems required to control the containment conditions in the event of a design basis accident. A steel containment has been avoided, due to a recommendation to avoid steel because a large number of operating nuclear power plants have corrosion issues. The recommended containment building will be constructed of reinforced and post-tensioned concrete with a metal liner. Since the SCWR thermodynamic cycle does not require steam generators, the size of the containment building is significantly smaller compared to PWR containments. Additionally, having the reactor building enclosed, the containment building will keep some of the components and systems in the reactor building outside the primary containment, which will improve access and inspection. A smaller primary containment will also reduce the construction cost. The containment building will be designed to withstand the pressures, thermal and mechanically-induced loads, and environmental conditions in case of a design basis event. The reactor building will protect the containment and the systems enveloped against external hazards. The reactor building will act as a secondary containment barrier and provide a means of collection and filtration of leaked fission products. Having a strong reactor building surrounding the containment increases the physical protection of the reactor and the containment.

The HPLWR is designed to confine the primary heat transport system as soon as the containment isolation valves are closed. The HPLWR containment has been designed to withstand an internal pressure increase up to 0.5 MPa, a horizontal ground acceleration of 0.8 m/s2 in case of an earthquake, and an airplane crash or an outside explosion pressure wave (Figure 30). The external event due to an airplane crash was decoupled by incorporating a gap between the reactor building and the containment, so the reactor building would sustain the mechanical load while leaving the containment structure on the inside unaffected by the impact.

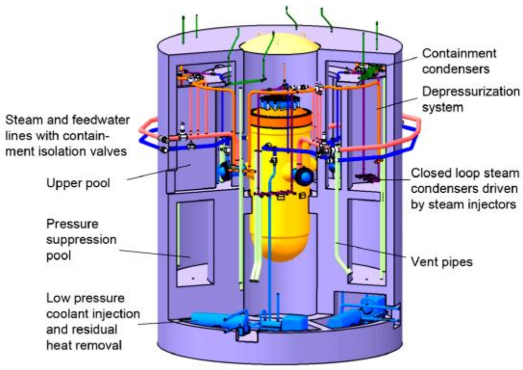

Figure 30: Arrangement of Safety Systems in the HPLWR Containment [6].

The HPLWR containment has an annular 900 m3 pressure suppression water pool with 500 m3 of nitrogen, four upper water pools of 1,121 m3 total capacity, and a drywell with a 2,131 m3 gas space. On the inside and outside there are four feedwater lines with check valves and four steam lines with containment isolation valves, each connecting the reactor to the steam turbine. The valves are able to close, actively or passively, within 3 s. There are four automatic depressurization systems, each equipped with two safety relief valves and two depressurization valves with a flow area of 110 cm2 each, connected to 8 spargers in the upper pools. The actuation pressure of the safety relief valves is set at 27.5 MPa.

Underneath the pressure suppression pool, there are four redundant and separate low pressure coolant injection pumps with 6 MPa outlet pressure and 180 kg/s maximum flow rate each. These supply coolant from the pressure suppression pool, via a heat exchanger for residual heat removal and a check valve, to the feedwater line. The overflow pipes from the upper pools to the pressure suppression pool complete the coolant loop inside the containment.

There are sixteen vent tubes from the pressure suppression pool to the drywell. Four emergency condensers are connected to the four steam lines and four feedwater lines. The flow through these condensers is driven by four steam injectors (Figure 31) to pump condensate flow into the feedwater line. Flow through the steam injector is initiated by opening a bleed valve to the spargers in the depressurization lines. As soon as subcooled liquid from the upper pool is sucked into the steam injector, the steam jet condenses inside the injector, which builds up a condensate pressure opening a check valve to the feedwater line. Once the flow has been established, the bleed valve is closed and the pressure vessel is depressurized slowly through the steam injectors, which build up enough pressure to refill condensate into the vessel and to drive a coolant flow through the core. The coolant mass flow is controlled by the control valve in the condensate loop. This system does not need power to drive a pump, but auxiliary power for the control system driving the valves is still needed.

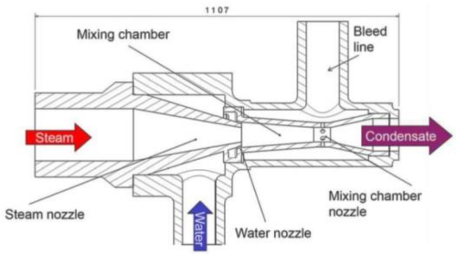

Figure 31: Conceptual Design of the HPLWR Steam Injector [6].

There are four containment condensers mounted on the ceiling of the drywell with their secondary side connected to pools above the containment. The secondary side is permanently open to steam in the containment and can condense as soon as the saturation temperature in the pools has been reached.

Open pipes from the ceiling to the pressure suppression pools enable hydrogen discharge from the drywell. The pressure suppression pool, in turn, can be vented to the stack through aerosol and iodine filters. Outside the containment, a boron poisoning system is located on top of the containment with a 10 m3 tank connected to the feedwater lines with pumps.

6.3.1 Materials

The SCWR components are subjected to high temperature gradients, high differential pressure and high through-wall stresses. The material selection for each component is dependent on: the strength requirements at relevant temperatures; corrosion resistance; fatigue behaviour; resistance to irradiation damage; creep resistance; and the cost of manufacturing. Preference is given to materials used in current reactors and high-temperature SCW fossil-fueled plants with proven performance experience.

The inlet plenum encases the outlet header, pressure tube extensions, and the high-pressure inlet coolant at 25 MPa and 350°C in the Canadian SCWR (Figure 7). Although the inlet plenum is a pressure vessel, none of the components are subject to high neutron fields and, consequently, irradiation damage is not a major concern. As a result, there is significant flexibility in material selection. A steel pressure-vessel containing approximately 3 wt.% to 4 wt.% nickel, SA 508 grade 4N, has been selected since the operating temperature inside the inlet plenum is only approximately 350°C.

The vessel and head are each manufactured from a single piece of forged material to avoid welds. The domed top of the inlet plenum is joined to the rest of the plenum at a bolted flange, which is sealed with a series of concentric metallic O-rings. Penetrations in this vessel exist for four inlet pipes and four outlet pipes. The inlet and outlet pipes are sized such that the inlet and outlet flow velocities are less than 7 m/s and 30 m/s, respectively. The Cr content of the materials is sufficient to prevent flow-accelerated corrosion (FAC) of these components. For the cold leg (inlet) piping, with an expected operating temperature up to 350°C, the reference material is chromium-steel P91 or P92 alloy.

The inlet pipe is located as high in the inlet plenum as possible so that, in the case of an inlet line break and depressurization, some coolant will remain in the inlet plenum providing cooling to the fuel channels. In addition, a vortex fluidic diode is located at the coolant inlet, which allows a low forward flow resistance and a much greater reverse flow resistance. This would limit the rate of depressurization on an inlet line break, allowing a greater time period of forward flow before depletion of the coolant inventory in the inlet plenum. It is recognized that the temperature difference between inlet and outlet streams is large and, hence, the resultant thermal stresses can be high in some components. This is addressed by insulating the outlet plenum. The inlet plenum and inlet/outlet piping are insulated to reduce heat losses to the containment.

The areas of maximum stress in the inlet plenum were computed for one complete start-up/shut down cycle [37]. High effective stress regions are: the top of the plenum head, the outer fillets of the outlet penetrations, the inner edge of the outlet penetrations, and the corner that joins the plenum to the tube sheet. As with other reactor concepts using large forged components, the Canadian SCWR concept relies on a limited number of ultra-large forging presses in operation globally to produce the inlet plenum and head components. The reference plenum weight is 600 tonnes, with a flange diameter of 7.4 m, which is the practical limit of today's presses. The head is considerably lighter at 198 tonnes with a flange diameter of 7.4 m.

In order to allow a bi-directional reversed re-entrant coolant flow path through the fuel channels, a separate vessel called the outlet header is placed within the inlet plenum, collecting the ~625°C SCW from the individual channels and directing this to one of four outlet pipes. The material selected for the outlet header and head is Alloy 800H, which is an iron, nickel, and chromium (Fe-Ni-Cr) alloy that demonstrates excellent high temperature properties such as strength, toughness, and corrosion resistance.

The Canadian SCWR concept uses the High Efficiency Channel (HEC) concept, which does not require a calandria tube to separate the pressure tube from the moderator; each pressure tube is in direct contact with the moderator. The coolant pressure of 25 MPa is transmitted through the metal liner and insulator, and applied directly to the pressure tube, which is the pressure boundary and must have high strength material to contain the coolant. In order to take advantage of the low neutron cross section of zirconium, yet allow seal welding of the pressure tube to the inlet plenum, the pressure tube transitions from stainless steel to zirconium at a point above the core, out of the neutron flux. This is achieved by means of co-extrusion, where a billet comprising of these two materials in the correct proportions and geometry is forced through an extrusion press die under a controlled atmosphere. In the extrusion process, the two materials are bonded at the molecular level, providing a joint whose mechanical properties are similar to the parent materials. The billet geometry is controlled in order to provide a tapered joint, whose length is many times the tube wall thickness. The pressure tube will be manufactured from a low-neutron absorbing zirconium alloy. The reference material for the pressure tube extension is Alloy 718 (UNS N07718), a precipitation-hardenable nickel-chromium alloy containing significant amounts of iron, niobium, and molybdenum along with lesser amounts of aluminium and titanium. A high-strength, creep-resistant zirconium alloy Excel (Zr - 3.5%Sn - 0.8%Nb - 0.8%Mo – 1130 part-per-million (ppm) O), developed by AECL, is the candidate material for the pressure tube of a Canadian SCWR concept.

Two candidate cladding materials, namely Alloy 625 and Alloy 800H, have been evaluated for the minimum and maximum cladding temperatures (426°C and 792°C), as determined by a thermalhydraulic assessment. For both materials the cladding is expected to collapse onto the pellet without forming a longitudinal ridge, for a cladding thickness range of 0.2 mm to 0.6 mm, while maintaining compressive hoop stresses in all of the cases. The results show the evolution of cladding collapse as the pressure increases, and the radial expansion of the cladding due to swelling of the fuel pellet, with increasing fuel burnup.

The low-pressure calandria vessel contains the heavy water moderator, fuel channels, reactivity control mechanisms, and emergency shutdown devices. Internal structures include lateral supports for the fuel channels, reactivity control mechanism guides, and flow channels ensuring circulation of the moderator. Currently, the calandria vessel pressure is set to 0.35 MPa and the operating temperature is approximately 100°C (40°C subcooled). As there are no significant changes in operating conditions in the Canadian SCWR concept compared to the CANDU reactor, the low pressure and low temperature calandria vessel should be manufactured from the same or similar stainless steel alloy as the calandria vessel for the CANDU reactor (e.g., Type 304L).

6.3.2 Safety barriers

The SCWR design, whether the HPLWR or the Canadian SCWR, uses multiple safety barriers. The first barrier is the retention of fission products in the nuclear fuel, by maintaining fuel temperatures below the melting points. A chemically stable fuel matrix is proposed. All accident analyses performed on the current designs ensured that the fuel centreline does not reach melting temperatures, thus retaining fission products within the fuel matrix, should environmental factors lead to sheath failure. Enclosure of the nuclear fuel in sheath/cladding tubes that can safely withstand the expected temperature range was determined by thermalhydraulic assessment. In addition, in the Canadian SCWR a no-core-melt concept has been incorporated to keep the fuel sheath temperature below its melting point through the use of radiation heat transfer during accident conditions. Further work is proceeding to find suitable accident-tolerant sheath materials for the required conditions. The fuel stack and the primary coolant are confined in a pressure tube (Canadian SCWR) or in a reactor pressure vessel (HPLWR). The material and thickness of the pressure tube/vessel wall of the reactor and primary circuit components are adequately designed to sustain the thermomechanical loads and chemical environments encountered during their design life. The pressure tube and the pressure vessels are being qualified for the pressure and temperature conditions expected in the SCWR concept. A double-walled reinforced-concrete containment structure, lined with an austenitic steel liner to form a hermetic seal, will contain the primary circuit components and pressure vessel/calandria. The primary containment is enveloped by a larger reactor building that is equipped with an ultimate heat sink (large water pools) as well as the safety-related mechanical components required to support and protect the reactor.

6.3.3 Source term

The source terms of radionuclides, for design basis accidents in Gen II and III reactors, consist of normal coolant radioactivity (tritium, noble gases and iodine mixture) discharged into containment through the break, plus fission products released from fuel elements/claddings that fail during the event. The number of fuel elements (pins) that are likely to fail during an event, and the timing of the failures, are established based on a detailed fuel behaviour analysis, assuming the worst thermalhydraulic conditions predicted for fuel element/claddings during the event. For analysis purposes, generally it is assumed that the whole inventory of the gap and 1% of the grain-boundary inventory are released from a fuel element at the time of sheath/cladding failure. Additional releases, from the fuel matrix, are considered only if the fuel temperature is predicted to stay high for a long period of time. Normally, this happens only if the Emergency Core Cooling (ECC) injection fails in the Gen II and III reactors. In the Canadian SCWR the gravity-driven core flooding system is an equivalent system to the ECC. The current design of the SCWR aims to prevent any source term release to the containment and thus the need for emergency measures and the evacuation of the public becomes redundant. In the HPLWR, the core is adequately cooled by four steam injectors to pump condensate flow into the feedwater line.

The gap inventory and therefore the source terms are not available for thorium-based fuel cycle using a mixture of thoria with reactor grade plutonium oxide. Additional R&D is being conducted to assess these quantities. The HPLWR fuel with 7% 235U enrichment also requires additional gap inventory assessment.

6.3.4 Containment bypass

Like in a BWR, the direct cycle SCWR has the potential for the main steam lines to provide a direct pathway between the reactor pressure vessel/tube and the environment. The technology used in the current fleet of reactors and the operational experience will be extensively used to improve the reliability of the containment isolation devices against containment isolation signals. In the SCWR, the main steam isolation valve will use a high reliability device to ensure containment is isolated during a failure of the main steam lines. Achieving a high reliability is feasible. Once the containment-isolation valves are closed, the next step is to ensure their leak tightness. This issue will be plant-specific because it is strongly dependent on local plant features, containment-isolation device design features, and management and maintenance issues. Although no severe accident is expected, the current containment isolation devices show high safety margins to withstand the pressure, temperature, radiation, and deformation loads resulting from severe accidents.

7. Management of design extension conditions (severe accidents)

The design extension conditions are accident conditions that are not considered for design basis accidents, but are considered in the design process of the facility in accordance with best estimate methods. For the design extension conditions, current practice expects that releases of radioactive material are kept within the acceptable limits stipulated by regulatory authorities.

The primary objective of the SCWR design concept was to achieve a higher state of safety compared with current Gen II and III designs. In the sections that preceded, analysis results were presented to the effect that: there is no possibility of re-criticality; the availability of continuous cooling of the fuel with an insignificant probability of forming core debris; and the no-core-melt concept preventing uncontrolled releases of radioactivity to the environment. All of these features potentially reduce the need for managing design extension conditions.

7.1 Prevention

The first level of protection expected from a SCWR design concept is the prevention of abnormal operation and system failures. Nuclear reactors are designed to ensure reliable, stable and easily manageable operation. Thus safety-critical components in a SCWR will be similar to current Gen III designs, incorporating very high-quality technologies. The prevention of abnormal operation will be based on highly-reliable instrumentation and control systems. The current CANDU system uses highly-reliable digital control systems; similar or better system will be incorporated to monitor pressure, temperature, coolant flow, neutron flux, and radiation to control criticality in a SCWR preventing an accident from escalating in severity beyond anticipated abnormal operation.

7.2 Mitigation

The mitigation of the consequences, subsequent to design extension conditions, is required to bring the reactor from a significant core-damage or core-melt state to a safe and stable state. The mitigating actions should be able to maintain this state indefinitely.

In the SCWR design, a number of control measures are available to arrest an abnormal-neutronic-power-and-fuel-cooling mismatch. There are two diverse and independent shutdown systems. The automatic depressurization system and low-pressure core-injection systems ensure continuous core cooling. In the Canadian SCWR design, a passive moderator cooling system ensures radiative cooling of the fuel to be more effective. The HPLWR design uses steam injectors to pump condensate into the feedwater line, to keep the core cooled without electric power.

The SCWR responds to pressure transients by using a pressure suppression design to vent overpressure through safety-relief valves, submerged below the surface of a pool of liquid water within the containment. The liquid pool is called the suppression pool and containment pool in the HPLWR and the Canadian SCWR designs, respectively. The Canadian SCWR design also uses atmospheric air heat exchangers to reject decay heat to the ambient air for long-term cooling of the core.

7.3 Situations to practically eliminate

The SCWR design practically eliminates external hazards affecting the shield building from propagating to the containment building, by placing a gap between the two structures. The design also practically eliminates core melt by using the radiative properties of fuel channel components. Additional work is being performed to demonstrate the feasibility of a full-scale fuel channel. Other R&D is being pursued to incorporate innovative solutions to practically eliminate initiating events that have the potential to disrupt core coolability.

8. Safety of the fuel cycle

8.1 Type of fuel

The fuel for the Canadian SCWR concept is similar to existing power reactor fuel, in that a ceramic pellet produces heat which is transferred through a metallic cladding to the primary coolant. Significant differences between the Canadian SCWR concept and existing power reactor fuels, which have been considered, are the normal operating conditions and accident conditions of higher temperature and pressure. These additional considerations (combined with corrosion concerns of SCW) necessitate the rejection of zirconium-based alloys as fuel cladding candidates. The reference fuel for the Canadian SCWR concept is a thoria-based fuel, with reactor grade plutonium (RGPu) recovered from used Light-Water Reactor (LWR) fuel as the initial contributor of fissile content in the fuel. The use of thoria-plutonia fuel in the Canadian SCWR concept presents some challenges in the development of the physics concept. The plutonium isotopic composition is based on that of sample SF97-4 [38]. For all of the fuel mixtures (Pu Th)O2 alone, and mixed with ErO3 or GdO3 it was assumed that the densities of the mixtures were simply the volume-weighted averages of the components. It was further assumed that the fuels in pellet form had densities equal to 97% of the theoretical density. The HPLWR is planning to use fresh fuel with 7% 235U enrichment and 2% Gd poisoning. Some combination of the fuel arrangements were discussed in Section 6.1.1.

8.2 Management of waste (quantity, quality)

The goal of SCWR conceptual design is to reduce waste mass, volume, thermal load on the repository, and the level of radiotoxicity. The reuse of plutonium from used HWR fuel or LWR fuel can reduce (between 25% and 50%) the decay heat and radiotoxicity for long term storage (i.e., from 1000 to 10 000 of years) [8], while the high-level waste per unit electricity produced is also reduced by approximately 25% when HWR Pu is recycled in the Canadian SCWR concept.

8.3 Radiation protection

The radiation protection is expected to follow the current Gen II and III levels, or exceed them over the total system lifetime, by adhering to applicable standards and regulations since the fuel types and coolants are not significantly different. The high-pressure turbine has been configured to minimize the material requirements and thermal stresses. It is also placed inside a separate containment for radiation protection. The concept of as-low-as-reasonably-achievable (ALARA) is expected to be employed. Since Gen IV nuclear energy systems must promote the highest levels of safety and reliability, a judicious pursuit of excellence in safety and reliability will be maintained.

9. Other risks

The manufacturing of 64 element fuel assemblies is a risk since it is a first-of-its-kind design with re-entrant flow. The chemical effects of SCW on fuel sheath, liners, pressure tube, and insulators require extensive testing and demonstration. Since SCW has not been used in a radiation environment as a coolant, the required understanding to adopt the fluid requires a significant level of experimentation and demonstration.

10. Summary of progress needed

Simplifying assumptions and extensions of existing data and methods have been applied in the development of the SCWR design concept. These assumptions and extensions require confirmation. The most significant gap in the fuel technology for the SCWR is considered to be the change in cladding material properties as a function of irradiation damage. Although there is reasonable confidence that in-reactor performance should be satisfactory, additional in-reactor R&D data is required.

The maximum diametral strain estimated for the Canadian SCWR pressure tube after 75 years of full power operation requires validation. This estimate was based on a very limited amount of data, so in-core irradiation experiments are required at SCWR conditions to validate this estimate. Additional experimental data on thermal conductivity, fuel qualification, and performance of (Th, Pu)O_2 is required for its implementation in SCWR.

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