Super-Critical Water-cooled Reactors (SCWR) Risk and Safety Assessment White Paper

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Abstract

This white paper was prepared by the Risk and Safety Working Group (RSWG), the Super-Critical Water-cooled Reactor (SCWR) System Steering Committee (SSC), and the Generation IV International Forum (GIF). It provides an overview of activities conducted by participating members in the application of the Integrated Safety Assessment Methodology (ISAM) developed by the RSWG. Based on the application of the ISAM, the SCWR SSC has established future Research and Development (R&D) needs for the conceptual design of SCWR systems. Several areas that require additional R&D and analysis have been identified to improve risk and safety performance of SCWRs. This paper provides a summary of the identified areas for improvement.

1. Introduction

The international nuclear community came together in the year 2000 and signed a charter in 2001 to create the Generation-IV International Forum (GIF) for collaborative research and development (R&D) on the next generation of nuclear energy systems. Risk mitigation and enhanced safety are two of the goals set for the Generation-IV (Gen-IV) systems development and serve as the basis to assess their performance. The Risk and Safety Working Group documented a set of five distinct analytical tools as an Integrated Safety Assessment Methodology (ISAM) to assess Gen-IV systems under development to ensure that safety is “built-in” rather than “added-on”.

The objective of this paper is to perform an assessment of the adequacy of safety provisions incorporated in the conceptual design completed so far for the Super-Critical Water-cooled Reactor (SCWR) systems. Improvements required to demonstrate defense-in-depth (DiD) and additional R&D as required are the expected outcome of this study. As design matures and proof-of-concept R&D outcomes become available, tools described in the ISAM methodology will be applied to advance the maturity of the design.

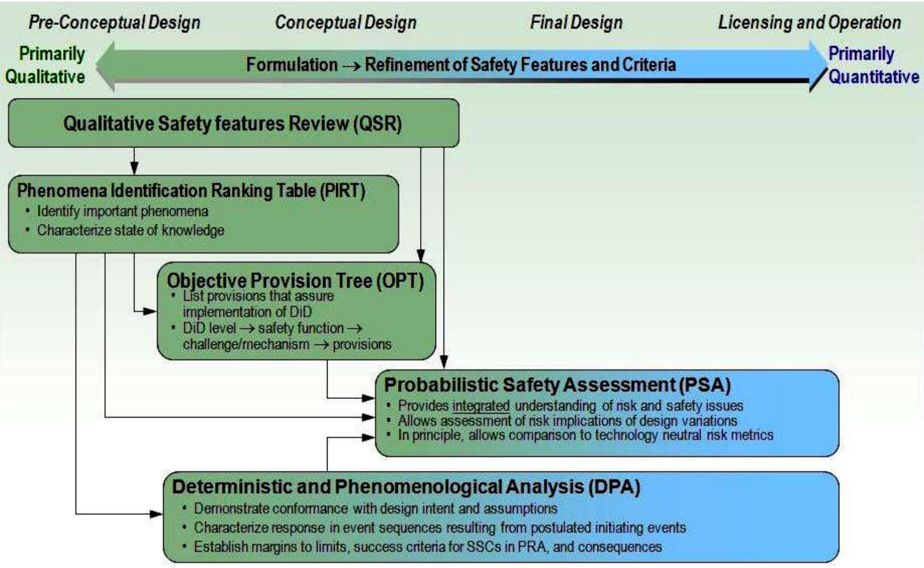
2. Review of the Integrated Safety Assessment Methodology

The Generation IV Risk and Safety Working Group (RSWG) developed an Integrated Safety Assessment Methodology (ISAM) to support the idea that safety is “built-in” rather than “added on” by influencing the direction of concept and design development. The methodology is useful throughout the Gen IV technology development cycle and consists of five distinct analytical tools[1]:

- Qualitative Safety Features Review (QSR)
- Phenomena Identification and Ranking Table (PIRT)
- Objective Provision Tree (OPT)
- Deterministic and Phenomenological Analyses (DPA)
- Probabilistic Safety Analysis (PSA)

Figure 1 shows the overall task flow of the ISAM and indicates the tools for use in each phase of Generation IV system technology development.

Figure 1: Proposed GIF Integrated Safety Assessment Methodology (ISAM) Task Flow[1]



Each of these analytical tools that is part of the ISAM is briefly described below:

- **Qualitative Safety Features Review (QSR)**

The Qualitative Safety Features Review (QSR) is a new tool of the ISAM. It provides a systematic means of ensuring and documenting that the evolving Gen IV system concept of design incorporates the desirable safety-related attributes and characteristics. Although this element of the ISAM is offered as an optional step, the QSR provides a useful means of shaping designers' approaches to their work and help ensuring that safety truly is "built-in and not added-onto" from the early phases of the design of Gen IV systems. Using a structured template to guide the process, concept and design developers are prompted to consider, for their respective systems, how the attributes of "defence in depth", high safety reliability, minimization of sensitivity to human error, and other important safety characteristics might best be incorporated. The QSR also serves as a useful preparatory step for other elements of the ISAM by promoting a richer understanding of the developing design in terms of safety issues or vulnerabilities that will be analyzed in-depth for other analytical steps.

- **Phenomena Identification and Ranking Table (PIRT)**

The Phenomena Identification and Ranking Table (PIRT) is a technique that has been widely applied in both nuclear and non-nuclear applications. As applied to Gen IV nuclear systems, the PIRT is used to identify a spectrum of safety-related phenomena or scenarios that could affect those systems, and to rank those phenomena or scenarios on the basis of their importance (often related to their potential consequences), and the state of knowledge related to associated phenomena (i.e., sources and magnitudes of phenomenological uncertainties).

The method relies heavily on expert elicitation, but provides a discipline for identifying those issues that will undergo more rigorous analysis using the other tools that comprise the ISAM. As such, the PIRT forms an input to both the Objective Provision Tree (OPT) analyses, and the Probabilistic Safety Analysis (PSA). It is particularly useful in defining the course of accident sequences and safety system success criteria. The PIRT is essential in identifying areas in which additional research may be useful to reduce uncertainties.

- **Objective Provision Tree (OPT)**

The Objective Provision Tree (OPT) is a relatively new analytical tool that has been increasingly accepted. The International Atomic Energy Agency (IAEA) has been a particularly influential developer and proponent of this analysis tool. The purpose of the OPT is to ensure and document the provision of essential "lines of protection" to ensure successful prevention, control or mitigation of phenomena that could potentially damage the nuclear system. There is a natural interface between the OPT and the PIRT in that the PIRT identifies phenomena and issues that could potentially be important to safety, and the OPT focuses on identifying design provisions intended to prevent, control, or mitigate the consequences of those phenomena.

- **Deterministic and Phenomenological Analyses (DPA)**

Classical deterministic and phenomenological analyses, including thermal-hydraulic analyses, computational fluid dynamics (CFD) analyses, reactor physics analyses, accident simulation, materials behaviour models, structural analysis models, and other similar analysis tools collectively constitute a vital part of the overall Gen IV ISAM. These traditional deterministic analyses are used as needed to understand a wide range of safety issues that guide concept and design development forming inputs into the PSA. These analyses typically involve the use of familiar deterministic safety analysis codes. It is anticipated that DPA will be used from the late portion of the pre-conceptual design phase through ultimate licensing and regulation of the Generation IV system.

- **Probabilistic Safety Analysis (PSA)**

Probabilistic Safety Analysis (PSA) is a widely accepted, integrative method that is rigorous, disciplined, and systematic, and therefore it forms the principal basis of the ISAM. PSA can only be meaningfully applied to a design that has reached a sufficient level of maturity and detail. Thus, PSA addresses licensing and regulatory concerns and is performed and iterated with a beginning in the late pre-conceptual design phase and continuing through to the final design stages. In fact, as the concept of the “living PSA” (one that is frequently updated to reflect changes in design, system configuration, and operating procedures) is becoming increasingly accepted, the RSWG advocates the idea of applying PSA at the earliest practical point in the design process, and continuing to use it as a key decision tool throughout the life of the plant or system. Although the other elements of the ISAM have a significant value as stand-alone analytical methods, their value is enhanced by the fact that they serve as useful tools in preparing for and shaping the PSA once the design has matured to a point where the PSA can be successfully applied.

Each tool is intended to be used in answering specific safety-related questions in diverse degrees of detail and during different stages of design maturity. This white paper presents an overview of the Super-Critical Water-cooled Reactor (SCWR) concept and describes the application of some of the ISAM tools and their outcome. As indicated in[1], the ISAM tools should be used throughout the concept development and design phases to derive insights to influence the course of the design evolution. The application of these tools would yield an objective understanding of risk contributors, effectiveness of safety-related design provisions, sources and impacts of uncertainties, and other safety-related issues that are important for a successful design. The tools also present a measure of design maturity, in terms of the level of safety and risk associated with the conceptual design relative to safety objectives.

This White Paper focuses on the safety of the Canadian SCWR and only a supplementary description is provided on the High Performance Light Water Reactor (HPLWR or European SCWR), because the HPLWR project ended before the first version of the ISAM became available. The ISAM methodology is expected to be applied in the next available European project on risk and safety of SCWRs.

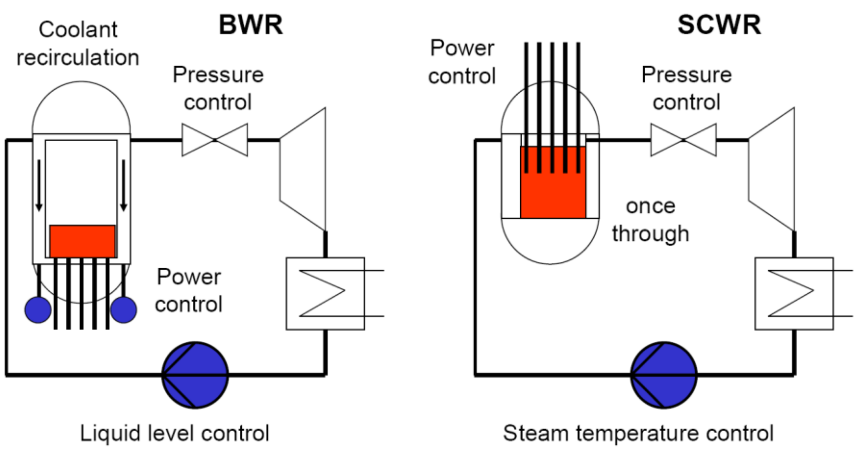
3. Historical Review and Lessons Learned from Past Reactor Construction and Operation Experiences

The SCWR concept is an extension of the trend set by the coal fired power plants in increasing the thermal efficiency and turbine power by increasing the live steam temperature. The increase to live steam temperature proceeded in unison with an increase to live steam pressure to maximize turbine enthalpy difference for a given condenser pressure. Coal fired power plants built in the sixties were designed with drum boilers, in which two-phase flow from the evaporator was separated in a drum, such that saturated steam was supplied to the superheater and liquid was recirculated either by natural convection or by a recirculation pump to the evaporator inlet. Since the nineties, the design of coal fired power plants started using higher live steam pressure, exceeding the critical pressure, which then annulled the need for steam and liquid to be separated any more. Consequently, the recirculation of liquid was omitted reducing the costs of the drum and the recirculation system. This simplified system is known as a once through boiler.

Boiling water reactors (BWRs) are still comparable to drum boilers with their separators and dryers and with their flow recirculation through the downcomer inside the reactor. An innovative approach to improving this system with a superheater was tried in the now decommissioned “superheated steam reactor” (Heiβdampfreaktor HDR) reactor in Grosswelzheim in Germany[2]. Unfortunately the HDR failed during commissioning phase due to a design error; however, an improved performance is expected from supercritical systems. Besides simplification of the system, supercritical water has the advantage of providing excellent heat transfer characteristics to the entire heat up range, from liquid to superheated steam, because phase change is not present avoiding the departure from nucleate boiling or dryout phenomena. Being a once through system, the separators, dryers, and downcomer fluid recirculation in a BWR are omitted in the SCWR. This latter simplification, however, causes a basic difference to the control of the steam cycle.

The simplified control systems of a BWR and a SCWR are compared in Figure 2. In a BWR, the feedwater pump controls the liquid level in the reactor pressure vessel, the turbine governor valve controls the steam pressure, and either the control rods or the speed of the recirculation pumps control the core power as shown in Figure 2 (left). The SCWR, shown on the right hand side of Figure 2, has no recirculation loop; however, the feedwater pump is able to either control the steam temperature at the core outlet if the core power is controlled by the control rods, or it can control the core power if the steam outlet temperature is controlled by the control rods. Again, the steam pressure in the SCWR is controlled by the turbine governor valve just as in the BWR case.

Figure 2: Comparison of BWR and SCWR General Features



The once-through system of the BWR (shown on the left of Figure 2) has an advantage in terms of general safety system requirements, because of having a closed coolant loop inside the reactor. It can remove residual heat by natural convection, driven by the rising steam in the core and above. The safety system has to ensure sufficient coolant inventory in the pressure vessel to keep the core covered with water. The SCWR (shown on the right of Figure 2), on the other hand, can remove the residual heat only by forced convection inside the reactor, which may be driven by a natural convection loop outside. The requirement for the safety system, in general, is ensuring sufficient coolant mass flow rate to remove the decay heat.

Besides this difference, there are several common safety system requirements, which can be taken directly from BWR concepts without significant modifications. These are:

- The reactor shutdown system consisting of either control rods or a boron injection system as a second and diverse shutdown system,
- Containment isolation by active and passive containment isolation valves in each line penetrating the containment to close the third barrier in case of an accident,
- Steam pressure limitation by pressure relief valves,
- Automatic depressurization of the steam lines into a pool inside the containment through spargers to close the coolant loop inside the containment in case of a need for containment isolation,
- A coolant injection system to refill coolant into the pressure vessel after intended or accidental coolant release into the containment,
- A pressure suppression pool to limit the pressure inside the containment in case of steam release inside the containment, and
- A residual heat removal system for long term cooling of the containment.

4. Overview of Canadian SCWR Technology

Based on the mechanical design of pressure retaining components of the reactor core, commercial nuclear reactors are grouped as either “pressure-vessel” or “pressure-tube” reactors. The most common pressure-vessel reactor designs are the Pressurized Water Reactor (PWR) and the Boiling Water Reactor (BWR). In these reactor designs, nuclear fuel is contained in a large pressure vessel. In pressure-tube reactor design, nuclear fuel is contained in a number of small pressurized fuel channels. The most common pressure-tube reactor used in commercial power plants is the Heavy Water Reactor (HWR), where common features are the fuel channel (including a pressure-tube, channel end closure and shield plug), a low-pressure heavy water moderator, a calandria vessel containing the moderator and fuel channels, and feeder pipes that transport coolant in and out of fuel channels. As a member of the Generation-IV International Forum (GIF), Canada is developing a pressure-tube-type SCWR concept that can potentially meet key technology goals of the GIF (i.e., improving economics and sustainability, as well as enhancing safety and proliferation resistance).

Increased pressure and temperature, needed to operate the reactor in supercritical water conditions, impose significant challenges to the reactor design. For example, under SCWR conditions, a typical reactor core material strength reduces by about a factor of two to three due to increased temperature (up to 625ºC) while the operating pressure increases by about a factor of two as compared to the current water-cooled reactors. The reduction in material strength and the increase in operating loads could be a limiting factor for the vessel size of a pressure-vessel type SCWR because of a need for a much thicker-walled vessel. The current manufacturing limit for a nuclear-grade SCW vessel size is about 4.5 metres in inner diameter with 0.5 m wall thickness.

On the other hand, a pressure-tube-type SCWR has no such manufacturing limitations or, if a pressure vessel is incorporated as in the proposed SCWR concept in this report, it is affected by much smaller extent as compared to the pressure-vessel-type SCWR. Today’s coal-fired supercritical power plants employ piping that can operate at SCW pressures of 25 MPa and temperatures higher than 650ºC, although the consequences can be widely different. This enables existing superalloy materials, thermal operating cycles, and turbines to be directly adapted to SCWRs.[3], [4]

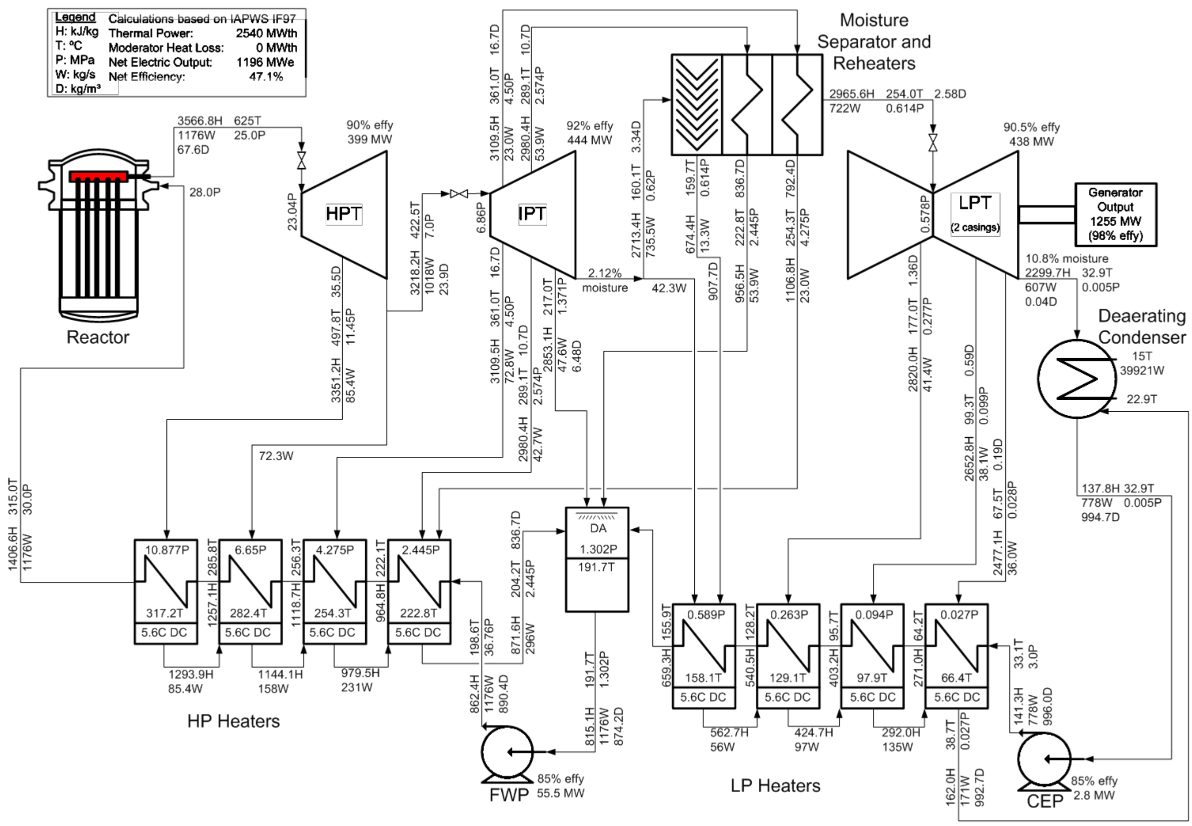
As in current designs, the pressure-tube-type SCWR uses a low-pressure calandria vessel and, as a result, many of today’s technologies (such as the control and safety systems) can readily be adopted with minor changes. Because the proposed concept uses a low-pressure moderator as in traditional HWRs, it does not require a pressure vessel that is subject to SCW pressures. Unlike current HWRs, the proposed concept uses batch refuelling, and to simplify the fuelling process, the reactor core is orientated vertically. Significant design simplifications result from the elimination of on-line fuelling systems, fuel channel end fittings and fuel channel.

The safety goal is based on achieving a passive "no core melt" configuration for the channels and core, so the mechanical features and systems directly reflect this desired attribute. Typical mechanical design requirements include meeting all applicable Codes and Standards, while being able to sustain a long operating and economic life with suitable allowance for maintenance compatible with decreased cost of generation.

4.1 Canadian SCWR Thermodynamic Cycle

The Canadian SCWR concept adopts a direct thermal cycle offering high efficiency and simplified system[5]. Figure 3 illustrates the typical SCWR layout which is similar to that of the boiling-water reactors (BWRs) and thermal cycle. The direct cycle directly passes steam into the high pressure turbines eliminating the need of steam generators as in the PWRs or PHWRs. Moisture separator and reheater (MSR) are used to remove water droplets prior to entering the low-pressure turbines. A 48% thermal efficiency can be achieved with the reactor outlet temperature of 625°C.

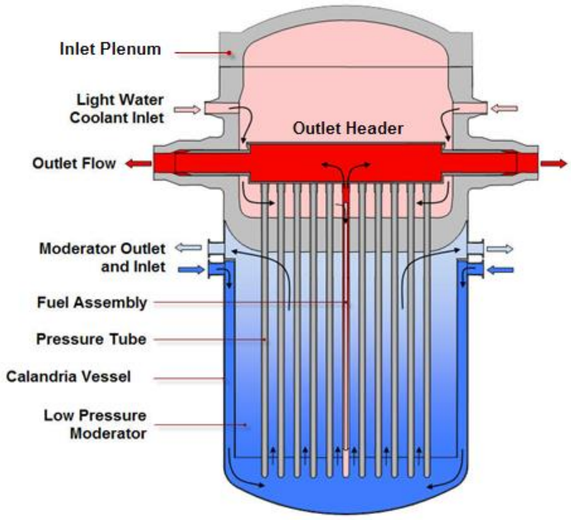
Figure 3: Schematic of Direct Steam Cycle with a Moisture Separator and Reheater in a SCWR Plant



The Canadian SCWR takes advantage of the well-developed balance of plant (BOP) layout of the SCW fossil-fire power (FFP) plants (which are operating at conditions similar to the Canadian SCWR). This would reduce significantly the design and development efforts. A noticeable difference in the BOP configuration between current SCW FFP plants and BWR plants is the implementation of the steam reheat feature, which eliminates the need of the MSR and increases the thermal efficiency further. This feature can also be implemented to a fuel-channel-type reactor. To match a SCWR to a reheat SCW turbine, the flow from the back end of the high pressure (HP) turbine must be returned at a lower pressure through the core in the second pass. The steam is then reheated to the required superheat and fed to the intermediate pressure (IP) section of the turbine. The reactor exit temperature can be established by either changing the channel length, the flow rate, or the number of passes through the core, or some combination. Reheat steam cycle version of the design uses channels to superheat steam which are placed at the periphery of the reactor core and have about 1.5 times lower heat flux compared to the average heat flux. Further development of the mechanical components is required to implement the steam reheat option, which has not been included in the reference Canadian SCWR concept.

A major safety feature of the Canadian SCWR is the passive cooling of fuel channels through the use of a flash-driven natural circulation of the heavy water moderator[5]. A schematic of the working principle is shown in Figure 4. Direct contact of the pressure tubes and the heavy water moderator keeps the pressure tubes cool during both normal and accident conditions. Following an accident, decay heat of the fuel is transferred from the cladding to the insulator through radiation heat transfer and then conducted to the pressure tube that is effectively cooled by the flash-driven naturally circulated moderator[6]. In the cold-leg large-break loss-of-coolant accident, which is the most limiting scenario, the peak cladding temperature is predicted to reach 1075°C, which is below the limiting cladding temperature 1290°C which is the lowest melting temperature of cladding alloy materials proposed for SCWR. The heat in the moderator is transferred to the reserve water pool and subsequently to the air (ultimate heat sink) heat exchanger outside of the reactor building. This continuous heat removal would ensure cooling of the fuel and maintaining the fuel and cladding integrities (i.e., no-core melt) long beyond the typical requirement of 72 hours (potentially indefinitely) without operator intervention.

Figure 4: Schematic and Flow Streams of the Canadian SCWR Core Concept



This increased safety feature satisfies one of the GIF goals of “improved safety” and this objective is a key constraint on the design concept in terms of channel design, bundle power and moderator heat removal.

While the overall BOP layout of the Canadian SCWR plant remains similar to that of the SCW Fossil-Fuel Power (FFP) plant, the high pressure (HP) and, possibly, the intermediate pressure (IP) turbines will be installed inside a separate containment to enhance the plant safety. Optimization of the piping layout is required, and further safety enhancement can be introduced using Dual Cycles, which utilize a steam generator or a heat exchanger, but with increased cost and loss of some thermal efficiency.

4.2 Canadian SCWR Core Concept

Traditionally, CANDU reactor cores have been placed horizontally, specifically to accommodate on-line refueling which maintains criticality while using natural uranium as fuel. Substantial work has already been done on developing horizontal SCW concepts because this was the natural evolution from the Generation II and III + channel HWR reactor designs (CANDU 6, PHWR and Advanced CANDU Reactor). For the channel type horizontal or vertical core SCWR concepts to be feasible, the high efficiency fuel channel (HEC), is preferred as it allows for fuelling from both ends of the core. Changes in fuel channel design and operating conditions necessitated modifications or new concepts in the conventional out-of-core fuel channel components interfacing with the HEC. Concepts of key fuel channel hardware such as a channel closure and the shield plugs were also developed at the concept stage. Details of the horizontal reactor concept and development are described in a paper on the development of SCWR out-of-core components[7].

Because of the challenge of the stresses and safety issues of connecting a fuelling machine to a pressure tube at SCW conditions, a vertical core design with off-power batch fuelling has been introduced[7]. In addition, the global move to a sustainable recycled fuel cycle using enriched fuels eliminates the necessity to use on-line fuelling, which is needed to maintain reactivity with natural uranium fuel. These considerations have led to the preference for the vertical orientation of the reactor core that allows more efficient batch refuelling as well as other design changes that simplified the reactor concept mechanical design.

Figure 5 illustrates the current vertical-core pressure-tube SCWR. The proposed design uses a pressurized inlet plenum attached to a traditional channel-type core. This design differs from traditional HWR designs in three major features: (1) using an inlet plenum instead of inlet feeders, (2) adopting a vertically oriented reactor core, and (3) refueled off-line. A cross-section of the reactor is shown in Figure 6 which provides a perspective of the entire reactor layout where the primary components are labelled.

In the vertical core the light water coolant enters the inlet plenum through inlet nozzles and then enters the fuel channels that are connected to the tubesheet at the bottom of the inlet plenum. Because the pressure losses in fuel channels are significantly larger than in the inlet plenum, inlet coolant will tend to divide reasonably uniformly into the fuel channels. However, further control of flow rates in each individual channel is needed with the appropriate use of orifices in fuel channels to obtain a more uniform exit temperature distribution. Inlet conditions are specified to be subcritical at a pressure of 25 MPa and a temperature of 350ºC[8]. As the coolant is forced vertically downwards in the fuel channel, it is gradually heated above the supercritical temperature with the energy generated from the fuel. The supercritical water exiting from the fuel channels is collected in the outlet header at an average 625ºC temperature chosen specifically to match existing and expected SCW turbines in thermal power plants.

4.2.1 Calandria Vessel

The tank holding the moderator surrounds the channels, and is conventionally called the calandria vessel (CV) and includes most standard CANDU features. The calandria vessel is a relatively low-pressure tank that includes heavy water moderator, the number of fuel channels needed for a given power, reactivity control mechanisms and emergency shutdown devices. The moderator is heavy water (D_2O) at low pressures and low temperatures because of its superior neutron moderation. The moderator operates slightly sub-cooled which makes it possible to use a flashing-driven natural circulation loop to passively remove moderator heat[9]. An attractive feature of this design is that this moderator system functions as a heat sink during both normal and off-normal operation and potential accident conditions, without active pumping, therefore, it provides additional safety margin and complete decay heat removal capability in case of emergency. The end shield at the bottom of the reactor is a neutron reflector filled with spherical steel balls as in current HWR designs. One of the design considerations for the calandria vessel is its ability to maintain structural integrity in the event of a sudden rupture of one or multiple pressure tubes.

4.2.2 Fuel Channels

A fuel channel consists of a small diameter pressure tube containing a nuclear fuel assembly and an extension linking the pressure tube to the outlet header. Because zirconium alloy material strength reduces sharply and corrosion rates increase at temperatures beyond 400ºC, the high temperatures in SCW fuel channels require special consideration. A ceramic insulator (placed between the fuel assembly and the pressure tube) is introduced to maintain the pressure tube temperature close to the moderator temperature[10]. The insulator material selected is Yttria-Stabilized Zirconia (YSZ), which has low neutron absorption properties and excellent thermal resistance. This fuel channel with zirconia insulator is called the High Efficiency Fuel Channel (HEC)[8] and is illustrated in Figure 7. Because of added insulator thickness and higher operating pressures, the HEC pressure tube is larger and thicker (as compared to current HWR reactors), and because the pressure tube is in contact with the moderator, this pressure boundary remains at the low moderator temperature. At this temperature, the zirconium alloy Excel has superior properties[11], [12] and, hence, is the selected material. The insulator is cladded between stainless-steel liners (Figure 8) that contain and isolate the insulator from the main coolant flow and fuel assembly. Since the physics, thermal hydraulics and mechanical aspects are coupled, fuel bundle optimization studies have been completed to ensure burn-up, enrichment, reactivity coefficients, axial and radial power profiles, fuel and clad temperatures, linear power rating, reactivity (k_{eff}), and stability are well behaved during the fuel cycle.

Figure 5: Schematic of the Canadian SCWR Core Concept

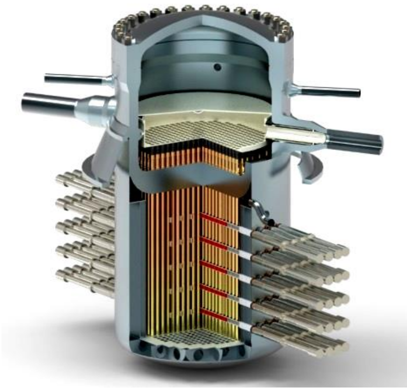
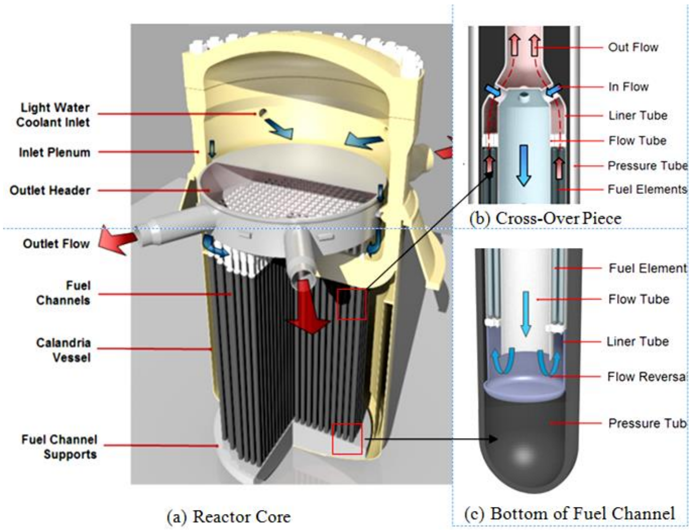
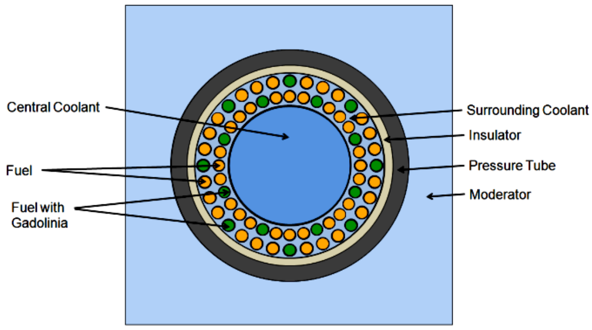


Figure 6: A Perspective View of the Reactor Layout with Labelled Primary Components



This insulated approach to the fuel channel has been adapted for several reasons, and is predicated on safety plus the performance limits of known materials under SCW conditions[13]. The various zirconium alloys are the only practical materials for use as pressure tubes because they have a low thermal neutron absorption cross section. Essentially all of the other possible structural materials that could contain the high pressures and temperatures of the coolant used in the heat transport system are based upon alloys of iron, nickel, cobalt, chrome or titanium. These alloys have much higher (factors of ten or larger in some cases) neutron absorption cross section that leads to significantly higher enrichment requirements for the fuel and, also, increases in activation products. However, the ultimate and tensile strength of zirconium alloys tends to drop quickly at temperatures of about 400°C. There is also a significant increase in strength as the temperature is lowered from around 300°C, typical of present use, to temperatures below 100°C.

Figure 7: A Cross-Sectional View of High Efficiency Fuel Channel with Zirconia Insulator



By using the pressure tube at low temperatures, one can utilize both the higher tensile properties and the lower neutron cross section of zirconium alloys. Currently, the moderator temperature is targeted to be ~80°C to maintain the pressure tube temperature below 100°C. The major passive safety feature is rejecting decay heat to the moderator without fuel melting, and direct contact of the pressure tubes and the heavy water moderator allows the natural circulation cooling of pressure tubes during both normal and accident conditions. Following an accident, radiation heat transfer from the fuel elements removes heat through the insulator from 750°C to the pressure tube and then via conduction and convection to the moderator, that is effectively cooled by the flash-driven naturally-circulated moderator.

The outlet end of the fuel channel is connected to the outlet header through a stainless-steel extension connecting to the pressure tube (Figure 9). The connection of the Excel pressure tube to the stainless-steel extension is manufactured using a co-extrusion technique, which has been confirmed at Chalk River Laboratories (further qualification of the process is required). An expansion bellow at the end of the fuel channel allows for axial thermal expansion and creep growth of the fuel channel. Outlet flow from the fuel channels is collected in the outlet header before it is transported to turbines.

Figure 8: The Thin Perforated Cylindrical Liner Tube Placed on Both Sides of the Zirconia Insulator

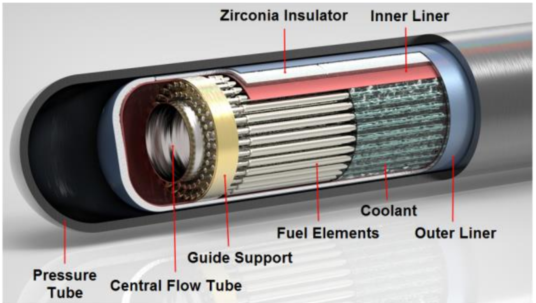
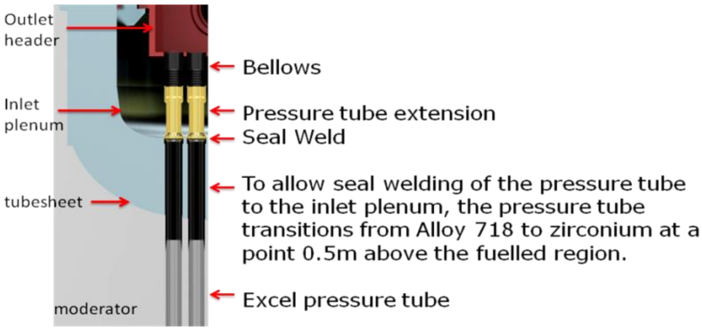


Figure 9: Fuel Channel Extension Configuration



4.2.3 Reactor Internals

The Canadian SCWR core is similar to the current CANDU design with the exception of the orientation. The current design of various reactor internals such as control rods and shutoff rods is similar to current CANDU technologies. Hence, these internals are inserted horizontally through the gaps between fuel channels (Figure 5). The control rods in a CANDU reactor operates in the low pressure moderator zone which prevents the potential for control rod ejection. The same feature is adopted in the SCWR. Because gravity-driven fail-safe mechanisms cannot be used as a result of the change in core orientation and avoid penetrating into a high pressure zone at the top of the reactor, other passive fail-safe mechanisms are used. There are three fail-safe approaches used in the Canadian SCWR: burnable neutron absorbers, adjuster rods and soluble neutron absorbers. As a result of this arrangement, no reactor internals are located inside the inlet plenum and outlet header.

4.3 Thermal Power

The thermal power of the concept can be easily varied to meet the need by varying the channel count, and has been set at 2520 MW for the nominal reference design. This has resulted in the electric power of about 1200 MW, assuming a 48% thermodynamic cycle efficiency of the plant. The resulting number of fuel channels is 336, obtained from the considerations of average fuel channel power of 7.5 MW(t), and a radial power profile factor of 1.2. The number of fuel channels is selected to be a multiple of 12 to allow flexible refuelling and shuffling the positions of 1/3rd of the fuel assemblies. Lattice pitch is selected to optimize fuel to moderator ratio to achieve a negative void coefficient and high fuel burnups.

5. Overview of the Safety Assessments of Canadian SCWR

The GIF Technology Roadmap identifies four technology goals for Gen IV nuclear energy systems: enhanced safety and reliability, improved economics, enhanced sustainability, and strengthen proliferation resistance and physical protection[14]. Both operational reliability and plant physical protection have been advancing from the Generation III and III+ nuclear systems, while the focus of improvement for future nuclear systems are mainly on economics, safety, sustainability, and proliferation resistance. In the post-Fukushima era, plant safety improvement has been the prime focus in securing public confidence and acceptance of nuclear energy.

In compliance to GIF ISAM, preliminary analyses using the ISAM tools of QSR, PIRT, and PSA were performed during the concept development phase of the Canadian SCWR. Strictly speaking, PSA is not required at the conceptual development phase. Nevertheless, a simplified PSA has been performed to quantify the core damage frequency (CDF) for the safety system of the Canadian SCWR concept.

5.1 Qualitative Safety Features Review (QSR)

QSR is the only tool in the ISAM specifically developed for Generation IV reactor systems. It is intended to provide a systematic means of ensuring and documenting that the evolving Generation IV system concept incorporates the desirable safety-related attributes and characteristics that are identified and discussed in[15].

A QSR for the Canadian SCWR concept has been performed. It is based on an exhaustive check list of safety good practices and recommendations applicable to Generation IV systems as suggested in[1]. These practices and recommendations for Defence-in-Depth (DiD) Levels 1~4 have been established. Design features and options are evaluated, identifying their strength or weakness, and qualified as Favourable (F), Unfavourable (U), Neutral (N), or Irrelevant (I) in relation with the desirable characteristics. Some of the design features evaluated are: reactor core, primary circuits, balance-of-plant, neutronic design, simple plant's thermo hydraulics design, heat removal at nominal operating conditions and during nominal operational transients, Decay Heat Removal, thermo-mechanical design (leak tightness, corrosion, vibration, thermo-mechanical loads during operational transients, etc.), postulated initiating events and mitigated consequences, and safety architecture. In case of loss-of-flow events, two independent shut down systems, an automatic depressurisation system, low pressure core injection system, passive moderator cooling system, and redundant feed water pumps are provided to arrest event progression. In this process, the characteristics and features are grouped in four classes, moving from general recommendations to detailed specific attributes:

- **Class 1**: Generic and technology neutral (applicable to all the technologies implemented in the innovative systems);
- **Class 2**: Detailed and technology neutral;
- **Class 3**: Detailed and technology neutral but applicable to a given safety function; and
- **Class 4**: Detailed and applicable to a given safety function and specific technology.

The QSR on Canadian SCWR was conducted during the concept development phase. From a total of 249 DiD Level-1 (i.e., prevention) features, 97 were assessed as “Favourable”, 41 as “Neutral”, 8 as “Irrelevant”, 2 as “Unfavourable”, and 101 as “to be assessed in the future”. It is worthy to note that two “unfavourable” features are associated with the direct cycle design of the system, so there is a potential that discharge will be ejected to the turbine hall rather than be confined within the containment. More work is required in the future to mitigate the consequences of those “unfavourable” features.

Some of the key features deemed favourable in the QSR on Canadian SCWR are:

- Fuel channel geometry and thermal properties optimized for decay heat removal following LOCA/LOECC so that the fuel sheath remains below the melting point,
- The frequency of postulated initiating events relating to positive reactivity transients, including the instantaneous removal of an adjuster rod (not physical but useful as a limiting case), and partial voiding of coolant have been minimized by ensuring coolant in the central flow tube remains,
- A reliable ultimate heat sink in the form of air coolers outside of the containment building, in addition to other water-based heat sinks,
- Plant design ensuring no leakage from the primary and secondary containments,
- Passive residual heat removal system to prevent unrealistic loss of heat sink events from occurring, and
- Redundant feedwater pumps to preclude inadvertent reduction of feedwater flow.

As expected, additional information and understanding of a number of phenomena are required because the Canadian SCWR concept is an innovative nuclear system with new phenomena that are not present in the current fleet of reactors (e.g., supercritical heat transfer and high temperature materials). The QSR assessment, especially that for features of DiD Levels 2, 3, and 4, will be finalized and formally documented in the detailed design phase of the Canadian SCWR, when more design details become available.

5.2 Phenomena Identification and Ranking Technique (PIRT)

The purpose of a PIRT process is to identify, recognize, and qualify the relative importance of all relevant phenomena with the associated rationales and their level of knowledge. The PIRT process consists of nine steps and allows the evaluation of a reactor concept or design by following how each phenomenon influences a key measurable parameter, called the Figure-of-Merit (FOM), chosen consensually by a panel of experts.

An application of the nine-step PIRT process to the Large-Break Loss of Coolant Accident (LBLOCA) for the Canadian SCWR concept was conducted. The selected accident scenario was a cold leg LBLOCA, which is the most limiting event compared to other accident scenarios. As expected, additional information and understanding of a number of phenomena are required because the concept is an innovative nuclear system with new phenomena that are not present in the current fleet of reactors (e.g., supercritical heat transfer and high temperature materials). A total of 428 phenomena were assessed for importance and knowledge level as summarized in Table 1. As expected, knowledge gaps were identified for 30 phenomena, which have “high” or "medium" importance rankings but have knowledge levels lower than "3". Among those 30 phenomena with knowledge gaps, 4 of them have "high" importance but with very limited knowledge (Level 1). These four phenomena are:

- Counter-current flooding limit in the central flow tube of the fuel assembly;
- Material degradation of ceramic insulator up to cladding failure;
- Material degradation of ceramic insulator after cladding failure to containment failure; and
- Cracking/embrittlement of ceramic insulator.

Additional information and understanding of those phenomena are required as in any innovative nuclear systems. Therefore, this PIRT will continuously be updated and formally documented in the detailed-design phase.

Table 1: Summary of Identified Numbers of Phenomena in PIRT

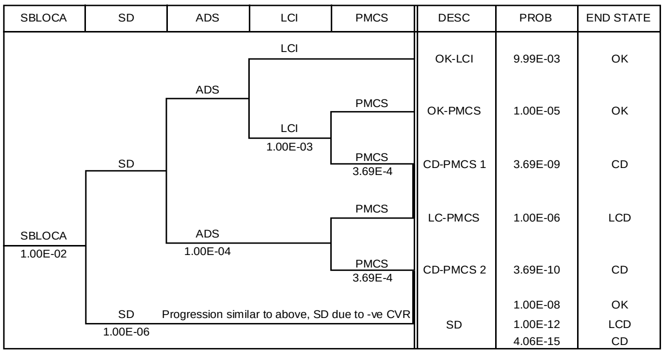
Rank Knowledge Level	Rank of Importance			
	H	M	L	I
4	3	2	1	10
3	67	37	4	242
2	25 (gap)	1 (gap)	8	21
1	4 (gap)	0 (gap)	0 (gap)	3

5.3. Probabilistic Safety Assessment (PSA)

A preliminary, simplified, PSA was performed on the safety system of the Canadian SCWR concept using the CAFTA (Computer Aided Fault Tree Analysis) tool[17]. The analysis covered the Small-Break Loss-Of Coolant Accident (SBLOCA), the Large-Break Loss-Of-Coolant Accident (LBLOCA) and the Loss-Of-Class-IV (LOCIV) Power event (which can lead to a Station Blackout Event). Comparisons were made to the response of the systems important to safety following the postulated accidents in the reactor. When an accident occurs, the systems responsible for controlling or stopping the accident are either initialized automatically or a signal is given to the operator so that the operator can manually stop the process. Details on the PSA methodology and data sources are described in[17]. The result of the event tree analysis for the SBLOCA is shown in Figure 10.

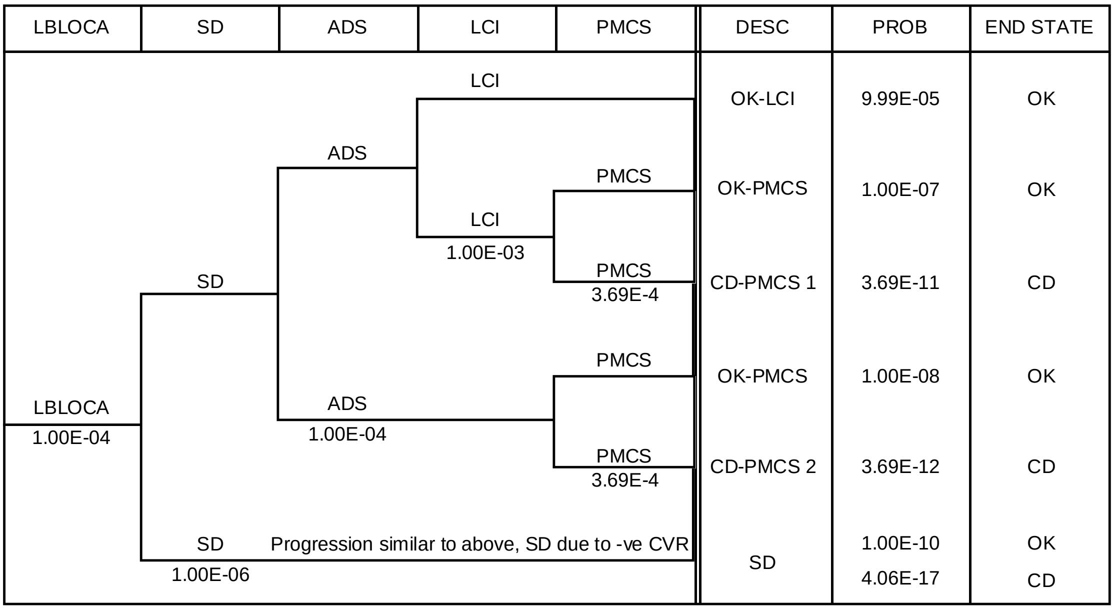
The terminologies applied in the analysis refers "SD" as the combination of the shutdown systems (SDS1 and SDS2) and the inherent neutronic characteristic of negative coolant void reactivity for the Canadian SCWR concept, "OK-LCI" implies that the core is cooled continuously (i.e., long-term cooling) with the Low-pressure Core Injection (LCI) system. If the LCI system fails, the core can still be kept cool with the Passive Moderator Cooling System (PMCS), which is referred to as "OK-PMCS". Limited Core Damage (LCD) may occur due to the reduction of cooling effectiveness when the Automatic Depressurization System (ADS) fails but the PMCS is still functioning. An LCD is defined as core damage occurring to a single channel that does not propagate to other channels. This is referred to as "LC-PMCS". Core damage could occur when the PMCS fails together with the failure of either the LCI system or ADS ("CD-PMCS1, CD-PMCS2).

Figure 10: Event Tree of SBLOCA in Canadian SCWR Concept



The result of the event tree analysis for the LBLOCA is shown in Figure 11. It is similar to that for the SBLOCA but the overall probability of core damage is much lower. The ADS is essential to maintain forward flow in the core and to ensure blowdown cooling until the LCI system becomes effective. In the event that ADS fails, the PMCS is the last line of defence to maintain continuous heat removal from the core. The introduction of the PMCS represents another level of defence-in-depth against the loss-of-coolant accident coupled with a loss-of-emergency core injection (LOCA-LOECI) event.

Figure 11: Event Tree of LBLOCA in Canadian SCWR Concept



The result of the event tree analysis for the LOCIV power event is shown in Figure 12. A LOCIV power event arises from failure modes, such as load rejection, loss of off-site power coupled with loss of station-generated power, a turbine or reactor trip, and a failure of the transfer busses to switch to grid supplied power. Coupling the LOCIV power with failures of the Class-III power supplies and emergency power could lead to a station-blackout event and subsequent loss of heat sink accident.

The analysis was much more complicated than those for SBLOCA or LBLOCA. It takes into the failure possibility of the following systems: the Reactor Regulation System (RRS), the Stand-By Generator (SBG), the Emergency Power System (EPS), the Isolation Condensers (IC), and the 72-hour power-restoration system, in addition to the ADS, LCI and PMCS. Overall, the PMCS could maintain cooling of the reactor even if other systems fail. Without the PMCS, on the other hand, limited core damage or core damage may be encountered but the probability of failures is much lower.

The probability of failures for the three postulated accident scenarios of the Canadian SCWR concept is summarized in Table 2. It is at least one-order of magnitude in improvement compared to other Gen-III reactor systems[17].

5.4 Summary of Preliminary Application

The method described in ISAM report was followed in applying the QSR, PIRT, and PSA tools to the Canadian SCWR concept. These three tools identified areas where additional knowledge is required to maintain the adequacy of safety-related system design.

6. Overview of the Safety Assessments of High Performance Light Water Reactor

The High Performance Light Water Reactor (HPLWR) is a pressure vessel type SCWR design. In this design, the minimum set of safety systems that fulfil ISAM requirements is shown in Figure 13. Reactivity control is provided by control rods and shut down rods, which deploy into the reactor from the top like in the pressurized water reactor (PWR), since there are no separators and dryers to complicate the design like in a BWR. An example of a HPLWR is given by Schulenberg and Starflinger[18]. In this design, the control rod drives outside of the reactor and the control rod guide tubes inside of the reactor can be duplicated from the PWR design without significant modifications. A vessel with boron acid must be provided inside or outside the containment for additional shut down under accidental conditions. Although different from PWR control, this boron acid cannot be used to compensate excess reactivity during normal operation; instead, some fuel rods doped with Gd as a burnable poison are required.

Figure 12: Event Tree of LOCIV Power Event in Canadian SCWR Concept

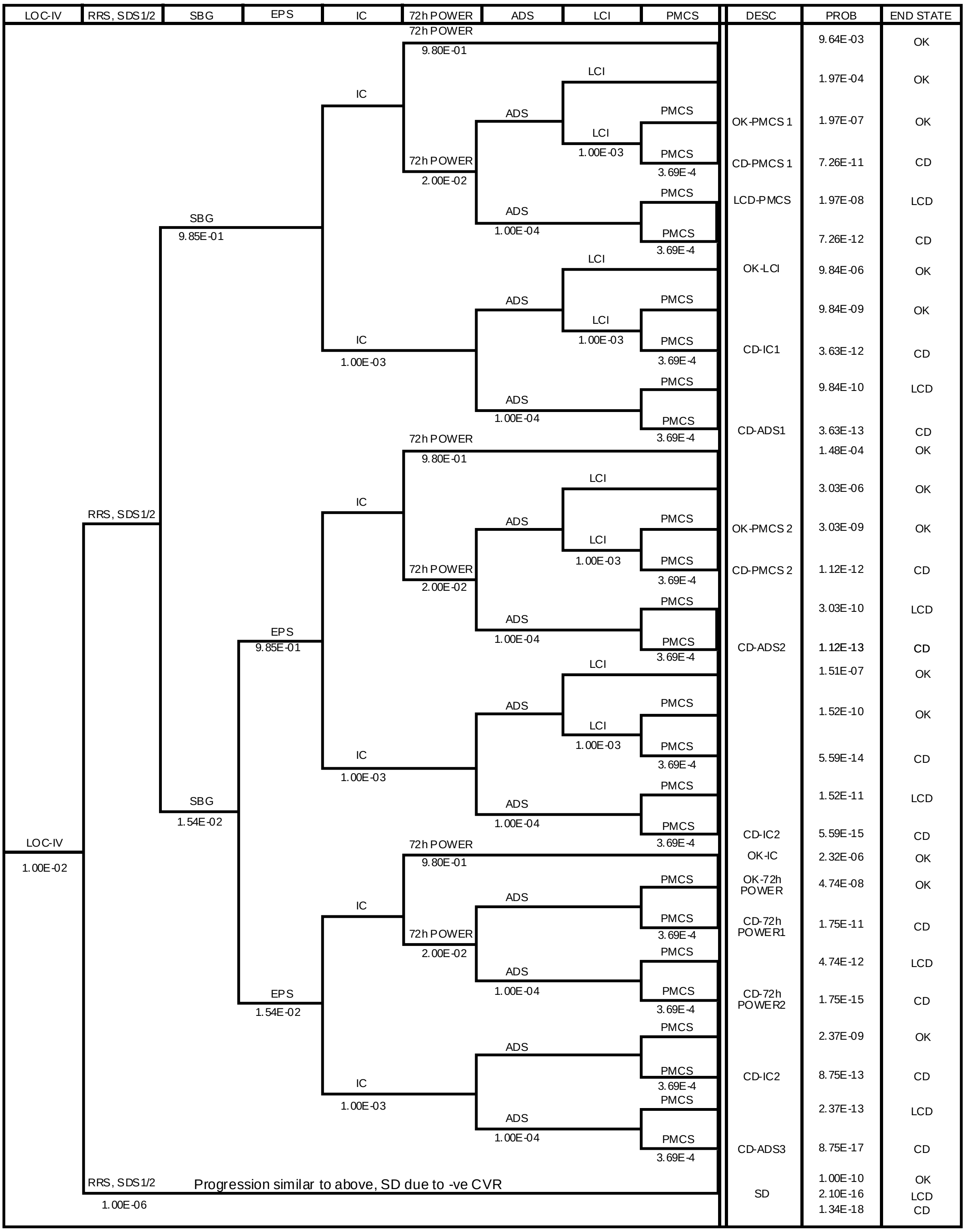
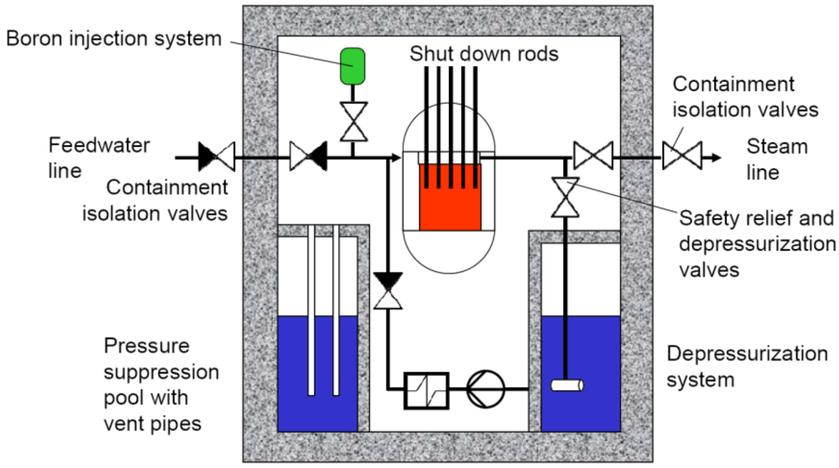


Table 2: Probability of Failures for Three Postulated Accident Scenarios

OUTCOME	Postulated Accident Scenarios		
	Small-Break LOCA	Large-Break LOCA	Loss of Class-IV Power
Cool Core	$1.00 \cdot 10^{-02}$	$1.00 \cdot 10^{-04}$	$1.00 \cdot 10^{-02}$
Limited Core Damage	$1.00 \cdot 10^{-06}$		$2.10 \cdot 10^{-08}$
Core Damage	$4.06 \cdot 10^{-09}$	$4.06 \cdot 10^{-11}$	$1.34 \cdot 10^{-10}$

Figure 13: Minimum Set of Safety Systems for SCWR



The containment isolation valves are check valves in feedwater lines, which need to be damped to avoid a water hammer, and steam isolation valves with hydraulic and medium controlled actuators. A pressure suppression pool with vent pipes maintains the containment pressure below the design limit, which can also serve as a heat sink for the automatic depressurization system in the simplest case. A low pressure coolant injection system with a heat exchanger to a secondary emergency coolant system is shown underneath the reactor in Figure 13, but it is meant to be placed somewhere inside or outside the containment in a sheltered position. These systems look quite similar as those of a conventional BWR. The different response of these systems in SCWR, however, shall be discussed by studying a loss of feedwater accident.

Let us imagine the case of a simultaneous trip of all feedwater pumps caused by a station black out. These feedwater pumps are high pressure, multistage centrifugal pumps which must be equipped with a check valve each to avoid backflow, in case of a trip of a single pump. These check valves, as well as those for containment isolation, will stop the feedwater flow within a few seconds and even potential flywheel inertia of the feedwater pumps cannot extend the short coast down time. Although this is different from a feedwater pump trip in a BWR, it is equivalent to a loss of coolant flow to the core occurring within a few seconds, requiring a scram of the reactor. As a consequence, the system must be depressurized immediately, being the only option to maintain a coolant mass flow rate through the core, either through the turbines and the turbine by-pass valves as an immediate action, or through an automatic depressurization system inside the containment to avoid loss of coolant outside of the containment. It is not wise to close the turbine governor valve in this case to keep a high system pressure, like in a BWR, as such measure would stop the steam flow simultaneously, which would overheat the core.

The pressure and coolant temperature history in case of containment isolation of all feedwater and steam lines has been simulated by Schlagenhaufer et al.[19] for the HPLWR with its coolant flow path as described by Schulenberg and Starflinger[18]. We see in Figure 14 (left) that the containment isolation will first cause a short pressure spike, which actuates the automatic depressurization system (ADS) of the steam lines, followed by rapid pressure decrease. The temperature history shown in Figure 14 (right) gives a short temperature spike, caused by a 0.6 s delay time of the ADS and 3.5 s shut down time of the control rods, but the coolant temperature falls rapidly afterwards to the feedwater temperature of 280°C because of the high coolant mass flow rate during depressurization. Within 20 s after containment isolation, the pressure reaches the saturation pressure of the feedwater inside the reactor and the feedwater in the upper plenum starts to boil. This situation will keep a minimum pressure in the vessel, initially at 6.4 MPa, which decreases slowly such that the core will be well cooled for about 10 min.

If the low pressure coolant injection system is designed with a pressure head of at least 6 MPa, and if the emergency power supply and the ramp up of the coolant injection pump can be provided within 20 s in total, the core will be well cooled for long term as the cooling circuit is now closed within the containment before significant amount of coolant is lost, and the residual heat is removed to the secondary coolant system. Although this time is short, but proven feasible in the conventional boiling water reactors. It would be advantageous, however, to have a longer grace period. The feedwater volume stored in the reactor pressure vessel has sometimes been called an “in-vessel accumulator”[20], suggesting that this water volume may be used for cooling before the low pressure coolant injection becomes available. Indeed, the core is well cooled by this water for about 10 min, as described above. As soon as cold water is injected, however, the steam pressure in the reactor breaks down and the core flow is stopped until the reactor has been filled up again. The in-vessel accumulator acts as a pressurizer, providing a driving pressure head for the coolant only as long as its temperature is sufficiently high.

Figure 14: Depressurization after Containment Isolation; Left: Pressure History, Right: Coolant Temperatures at Different Locations of the HPLWR Core Concept[19]

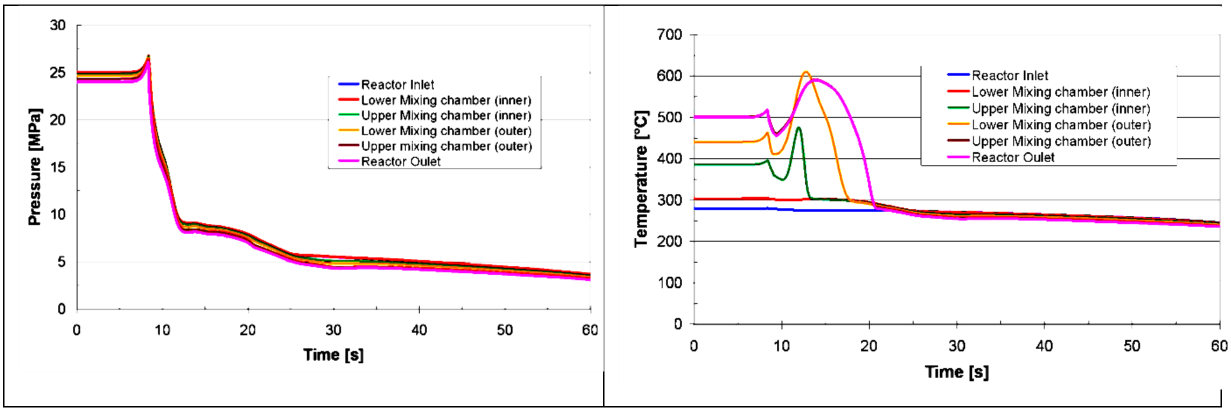
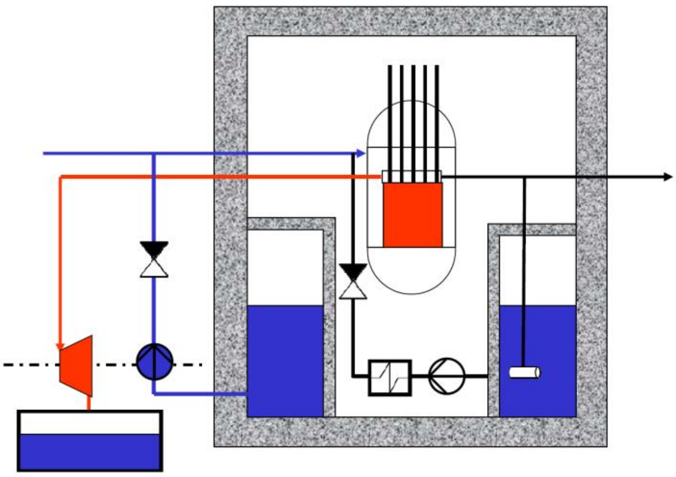


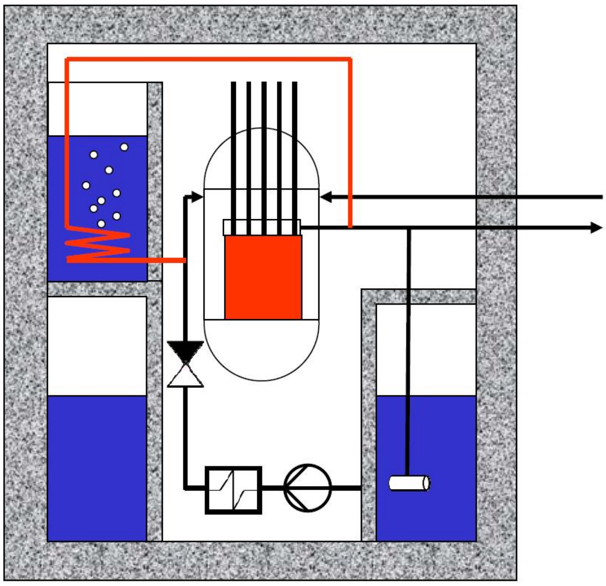
Figure 15: Depressurization through a Steam Turbine Driving a High Pressure Coolant Injection Pump



The problem can be overcome if the system is depressurized through a steam turbine driving a high pressure coolant injection pump, as sketched in Figure 15. Such system has often been used already in conventional BWR. As the condenser behind this turbine must be at lower elevation than the turbine outlet, but the pump intake must be lower than the water reservoir, this concept is usually designed with 2 coolant pools at different elevation. As sketched in Figure 15, Ishiwatari et al.[21] propose to use a separate condensate storage tank at lower elevation, like in a BWR. Now the missing coolant will be refilled already during depressurization. The steam mass flow must be high enough to ensure that the maximum cladding surface temperature in the core does not exceed the envisaged limit, but small enough to maximize the grace period for the active, low pressure coolant injection system.

A passive system without rotating components could be a closed loop which condenses the steam in an additional elevated pool inside the containment, as sketched in Figure 16. This system decreases the system pressure slowly, but the flow rate could eventually be too small to cool the core. Therefore, Schulenberg and Starflinger[18] propose to drive the coolant loop additionally with a steam injector. After an initial short depressurization through the ADS, the subcritical steam is supplied to the steam injector which drives a closed coolant loop through the condenser in the elevated pool. Coolant is lost to the containment pool only during the short initial depressurization phase, and the steam supplied to the injector afterwards will condense in the closed system. This innovative system, however, has never been analyzed in detail yet.

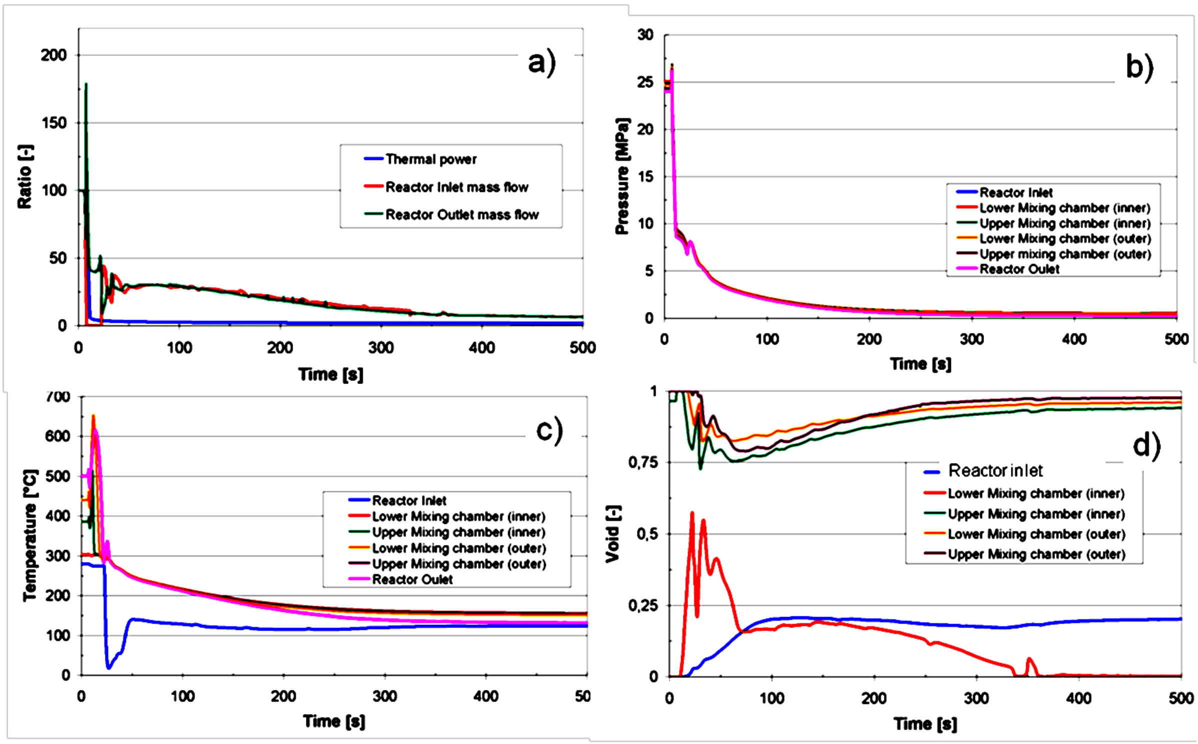
Figure 16: Depressurization by condensation in a closed loop inside an upper containment pool



Instead of the above mentioned system, Schlagenhaufer et al.[19] propose to use a motor driven recirculation pump to drive the closed coolant loop through the condenser in the elevated pool, which is easier to control and thus easier to optimize for this purpose. Figure 17 shows the response of the reactor coolant during a depressurization transient using this system. The transient was initiated at 5 s by inadvertent containment isolation, which caused a pressure peak, scram and activation of the ADS as described above. Despite the peak mass flow rate of the steam leaving the reactor outlet (a), a short temperature peak of the coolant cannot be avoided again (b). Simultaneous with scram activation, a recirculation pump in the condensate line of the closed loop is started, and the ADS is closed again as soon as the pressure is less than 10 MPa. As a consequence, the coolant temperature at reactor inlet drops suddenly to 20°C at 20 s, as the condensate stored in the loop has been cold during normal operation. Around 1 min after scram, the closed loop has stabilized, and the condensate temperature increased to the saturation temperature of actual system pressure (c). The core outlet temperature is controlled by the recirculation pump such that it stays slightly superheated, which minimizes the coolant mass flow rate and thus the required power of the pump.

Figure 17 (d) shows more than 90% void in the core after 5 min and even the feedwater inside the reactor "in-vessel accumulator" is calculated to be boiling, but the core remains cooled sufficiently. The total peak power of 4 recirculation pumps needed for this system is 1 MW. This exercise could serve as a starting point for a passive system, e.g. with a condenser at a higher elevation.

Figure 17: History of Core Power and Coolant Mass Flows (a), Coolant Pressure (b), Temperature (c) and Coolant void (d) during Depressurization in a Closed Loop[19]



The potential arrangement of such safety systems inside the HPLWR containment is illustrated in Figure 18. The active low pressure coolant injection system and its heat exchangers for residual heat removal are placed underneath the annular pressure suppression pool. A condensate recirculation pump of a closed loop, as described above, would need to be installed to have enough pressure head at its intake to avoid cavitation. Four elevated pools serve as a heat sink for the ADS and for the closed loop condenser sketched in Figure 16. Containment condensers hanging from the ceiling of the containment transfer residual heat to the pools above the containment when the temperature of the four elevated pools and of the pressure suppression pool has reached the saturation temperature, limiting the containment pressure. With 26.7 m total height and 21.6 m outer diameter, the containment is surprisingly compact, indicating a significant potential for cost reduction, but it is storing more than 2000 m^3 of water.

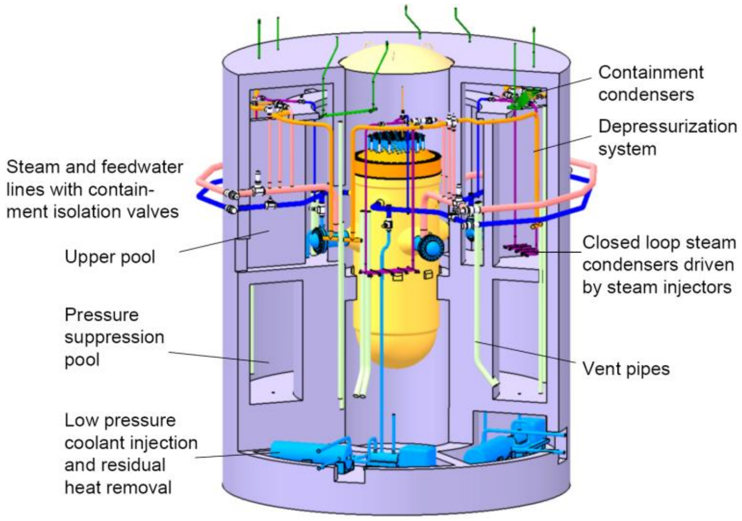
7. Severe Accidents

Despite drastically reducing the probability of severe accidents in SCWR conceptual design, mitigative measures will be incorporated to avoid short- or long-term post-accident escalation of the accident requiring evacuation of population around the reactor site boundary.

8. Conclusions

Conceptual supercritical water cooled pressure-tube and pressure vessel reactor designs are presented. The proposed pressure-tube reactor concept includes an inlet plenum, a tubesheet, a low-pressure calandria vessel, the high-efficiency fuel channel containing the fuel assembly and an outlet header. The fuel channel consists of an Excel pressure tube and a stainless-steel extension connecting the pressure tube to the tube sheet. The fuel assembly includes a 5-m long fuel bundle housed inside a sealed insulator.

Figure 18: Arrangement of Safety Systems in the HPLWR containment



When compared to present (Gen III) pressure-tube based CANDU reactors, the present SCWR concept offers the following advantages in mechanical design and operations:

- Eliminates inlet feeders,
- Eliminates channel closure plugs (two for each fuel channel),
- Eliminates channel closure seal (two for each fuel channel),
- Allows batch fuelling with simultaneous multi-channel fuelling and convenient access through a hollow inlet plenum,
- Easier pressure tube replacement with convenient access to fuel channels by removing the inlet plenum head,
- Enables a compelling safety case,
- Passive core cooling is made possible through natural convection of heavy water moderator,
- Small LOCA of an outlet has a small impact due to the common coolant inlet,
- Provides 40% higher efficiency, and
- Presents a compact footprint and layout.

So far, the design work included concept development and qualitative evaluation of design concepts supported by simple calculations. Detailed structural, thermal and hydraulic analyses are needed to evaluate the feasibility of the proposed concepts. Through numerical analyses and test programs, feasibility of the concepts will be demonstrated. Where deficiencies and issues are identified, modifications to proposed solutions should be made or alternate concepts should be developed. At the first stage, the following tasks are identified. It is anticipated that, while performing these tasks, many more tasks will be identified.

Safety analyses of the Canadian SCWR concept and the HPLWR concept covered key accident scenarios expected during the operation. The LBLOCA remains the limiting scenario with a maximum predicted cladding temperature of 1075°C for the Canadian SCWR concept.

There are gaps in understanding how the reactor core will behave under SCWR pressures and temperatures. The proposed core concept requires many engineering assessments to evaluate the structural integrity and heat transfer characteristics. The results of these analyses will provide feedback to refine the core design concept.

ISAM was not implemented from the start of the conceptual design but ISAM components of QSR, PIRT and PSA were applied once the design had progressed to that level that the application of these elements became feasible. Following the application of QSR two features were identified as “Unfavourable”, which were associated with the direct cycle configuration of the Canadian SCWR concept. These two features identified that there is a possibility of discharging fission products into the turbine hall rather than being confined within the containment. An indirect steam cycle can be included as the mitigation option and will be determined at the optimization phase. More work is required in the future to mitigate the consequences of these “unfavourable” features. The structured QSR exercise pointed towards the need for additional information and improvements to understanding on a number of phenomena. This is an expected outcome because the Canadian SCWR concept is an innovative nuclear system with a few new phenomena that are not present in the current fleet of reactors (e.g., supercritical heat transfer and high temperature materials).

The application of PIRT identified knowledge gaps in four phenomena with “high” importance and very limited knowledge:

- Counter Current Flooding Limit in the central flow tube of the fuel assembly;
- Material degradation of ceramic insulator up to cladding failure;
- Material degradation of ceramic insulator after cladding failure to containment failure; and
- Cracking/embrittlement of ceramic insulator.

The identification of these phenomena indicated that the novelty of the design is hinged on R&D demonstrating beyond doubt that there are no unforeseen behaviours within the expected range of operating, accident, and ageing conditions. The lessons from this exercise demonstrated the benefit to performing PIRT reviews periodically, updating and formally documenting as the design matures.

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