Research and Development Initiatives in Support of the Conceptual Design for the CANDU Supercritical Water-Cooled Reactor

D. Brady[*], D. Guzonas[†], W. Zheng[‡] and L.K.H. Leung[†]
[*] Office of Energy Research and Development, Natural Resources Canada, Ottawa, ON K1A 0E4
[†] Chalk River Laboratories, Atomic Energy of Canada Limited, Chalk River, ON, K0J 1J0
[‡] Material Technology Laboratories, Natural Resources Canada, Ottawa, ON K1A 0G1

Abstract

Canada's Generation-IV National Program has been established to provide research and development (R&D) supports for the concept of the CANDU®[1] SuperCritical Water-cooled Reactor (SCWR). It focuses primarily on the key technology areas, such as material, chemistry, thermal-hydraulics, safety, physics, and hydrogen production. Design challenges surrounding out-of-core components are also examined. R&D results of these areas will be applied in finalizing the design options currently being considered in developing Canada's conceptual design. In this paper, a brief description of the conceptual design options is presented and the linkage of the R&D projects to support the decision in selecting the reference design option for the CANDU SCWR is described. Various participants contributing the R&D information to the Gen-IV National Program are introduced.

1. Introduction

Canada is a founding member of the Generation-IV International Forum (GIF) and is actively undertaking research and development (R&D) in support of two of the six Generation-IV (Gen-IV) system designs (i.e., Very High-Temperature Reactors (VHTR) and SuperCritical Water-cooled Reactor (SCWR)). The Canadian effort focuses mainly on a pressure-tube type SCWR design, which is viewed as a natural extension of the current fleet of CANDU reactors. Canadian VHTR R&D is mainly focused on areas of high-temperature materials and hydrogen production, which are synergistic with the pressure-tube SCWR design. Several critical research areas have been identified in the System-Research Plan (SRP) for supporting the SCWR design: design and integration, chemistry and material, and thermalhydraulics and safety.

Based on preliminary design specifications highlighted in the GIF SCWR SRP and a preliminary Canadian concept, key research projects have been identified in the program covering material, chemistry, thermalhydraulics, safety, physics, and component design. Results from these projects have led to improvement and refinement of the conceptual design options. The effort in the first several years has also led to the identification of research gaps, especially in the area of materials and chemistry. The objective of this paper is to present some preliminary R&D results and their impact on the conceptual design options for the CANDU SCWR.

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

2. Canada Gen-IV National Program

The Canadian Gen-IV National Program was initiated in 2006 for Canada to support its international obligation under the GIF Framework Agreement (Treaty). The initial R&D undertaken was aimed at supporting the development of a conceptual CANDU SCWR design and included participation from various federal laboratories: Natural Resources Canada (NRCan) Material Technology Laboratory (MTL), National Research Council (NRC) of Canada and Atomic Energy of Canada Limited (AECL) [1].

It has been the vision of the National Program to have a strong university component due to the long-term nature of the R&D requirements. To this end, NRCan, along with the Natural Sciences and Engineering Research Council of Canada (NSERC) and AECL established the NSERC/NRCan/AECL Generation IV Energy Technologies Program, focusing on key areas of research for the SCWR system. A targeted call for proposals was issued to the universities in 2008 through the NSERC/NRCan/AECL Generation IV Energy Technologies Program. In the winter of 2009, 23 university projects were selected to be part of the NSERC/NRCan/AECL Energy Technologies Collaborative R&D program [2].

In addition to directly funding projects under the Gen-IV National Program, the program seeks to leverage and coordinate its funding with other national initiatives, such as the AECL CANDU-X program and the facility expansion of MTL as part of its relocation to Hamilton; these become in-kind contributions to the national program and thus offered as part of Canada's international contribution.

3. CANDU SCWR Design Option

AECL has been focusing on the development of a pre-conceptual CANDU SCWR design based upon preliminary R&D results obtained to date. The preliminary concept driving the R&D program was based on leveraging the existing CANDU reactor design configuration with horizontal fuel channels and online-refuelling capability. A project initiated to examine out-of-core component designs (such as channel closures and feeders) revealed that while extending various out-of-core components to SCW conditions (i.e., 25 MPa @ 625°C) is feasible, considerable challenges remain, particularly for the design of the online-refuelling machine [3].

Thorium-based fuel is also being considered as a potential reference fuel to enhance sustainability and proliferation resistance and physical protection (PRPP) of the CANDU SCWR, better aligning the CANDU SCWR to the GIF Technology Goals. In view of the need to use low-enriched uranium (LEU) or plutonium (Pu)/thorium fuel, the burn-up of LEU(or Pu)-thorium fuel will be much higher than that of natural uranium fuel. Therefore, implementation of online refuelling is no longer essential and batch refuelling is considered an acceptable alternative approach. The sequence for batch refuelling is more efficient for a long fuel string in a vertical fuel-channel orientation than multiple short fuel bundles in a horizontal orientation. Given the challenges of on-line refuelling when the coolant is pressurized at 25 MPa and the well-established use in Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs), a vertical core option seems a reasonable approach. Other advantages for adopting the vertical channel orientation are the improvement in heat-transfer characteristics (details to be provided in Section 4.3) as well as the elimination of pressure tube (PT) sagging due to high-temperature creep of in-core materials.

The low-pressure calandria design is similar to the current CANDU reactor in which the calandria vessel pressure is close to atmospheric and fuel channels are maintained in a modular configuration. An inlet plenum is introduced to simplify the channel entrance configuration; it facilitates the elimination of individual channel closures, which need to be quite large to withstand the high pressure inside the channel. It also eliminates the need for inlet feeder pipes, and the accompanying corrosion and corrosion product transport issues. Finally, the plenum provides the flexibility to optimize the lattice pitch between channels to meet the requirement for coolant void reactivity.

Introducing a plenum design at the channel outlet is challenging due to the high pressure and high temperature environment (i.e., 25 MPa and 625°C). Since refueling will be carried out at the inlet end, a channel closure is no longer required at the outlet. The feeder pipe is connected directly to the channel outlet, which in turn connects to one of several header pipes. A smooth transition is designed between the channel outlet and the feeder pipe. Header pipes linking all feeder pipes are directed to the outlet header. Figure 1 illustrates the low-pressure calandria design concept of the CANDU SCWR [3].

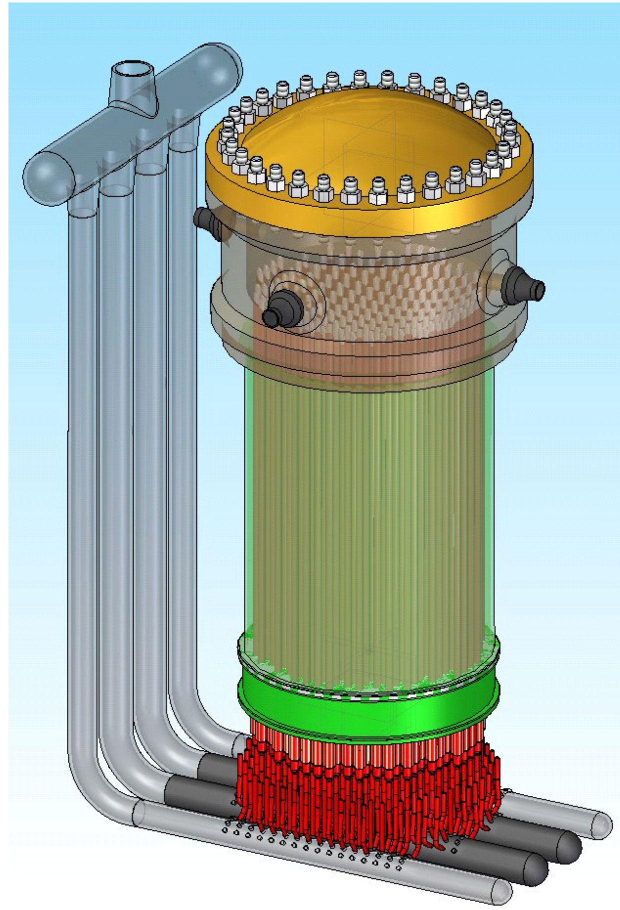


Figure 1 Schematic Diagram of One Design Concept of CANDU SCWR

In this concept, the reactor core consists of 300 channels, each with a nominal 5-m active length. Work is currently underway to optimize the fuel bundle design based on physics, thermalhydraulics, and fuel calculations.

4. CANDU-SCWR Research

Canada is engaged in all research areas related to the development of the SCWR, and the National Program builds upon expertise in both fossil-fired supercritical water (SCW) power plants and pressure-tube nuclear reactor design [4]. In order to support the viability of the CANDU SCWR and ensure GIF technologies goals are met, research is currently focused on four key technology areas: material, chemistry, thermalhydraulics, and safety.

4.1 Material Research Projects

A major focus of the Canadian program addresses critical SCWR materials challenges that arise because of the high temperatures and potentially very oxidizing environment in-core and in piping immediately downstream of the core. Materials research is currently focused on four key areas [2]:

1. Assessment of available alloys and development of new alloys for SCWR applications
2. Development or modification of surface coating processes for producing corrosion and SCC-resistant materials
3. Assessment of Excel Zr alloy as a candidate pressure tube material
4. Mechanistic studies of the fundamental corrosion processes of alloys in this special (partially water-like and partially gas-like) state of water.

While most of the work underway is experimental, some work is also being performed to model the effect of radiation damage on materials [4], and focussed modelling may be a valuable tool to assess potential candidate alloys under in-core conditions as high flux irradiation study on materials is constrained by the limited availability of test facilities.

Various coatings have been considered for use in the primary coolant systems of water-cooled reactors; for example, Kim and Andresen [5] proposed the use of yttria-stabilized zirconia coatings on Boiling Water Reactor piping to minimize inter-granular stress corrosion cracking of sensitized stainless steel, by reducing the oxygen transport rate to the surface. Deposition techniques included hydrothermal deposition, plasma spray, and sol-gel methods. Guzonas et al. [6] proposed the use of zirconia coatings, formed by a sol-gel process, on carbon and stainless steels to reduce the corrosion rate in SCW, and Wills et al. [7] examined the use of an atmospheric pressure plasma jet to deposit corrosion-resistant coatings for use in SCW. The stability of ceramics in SCW has recently been reviewed by Sun et al. [8]. Recent work has shown that spray pyrolysis coating of P91 substrates can reduce the corrosion rate by factors of 2 to 10, and a plasma electrolytic oxidation process has shown some promise for reducing corrosion of Zircaloy in SCW [9]. The challenges in this area include compatibility of the coating to metal substrate, mismatch in thermal expansion characteristics, and long-term metallic to non-metallic interface bond strength.

Given the difficulties in finding a single alloy with both the required bulk and surface properties, a duo-material surface alloying approach [10] is being investigated. The basis of this approach is that a corrosion-resistant surface layer can be produced on top of a substrate material having strong bulk properties and that an intimate metallurgical bond between them can ensure interface adhesion and long-term mechanical performance. Various surface alloying methods can be explored to fabricate in-core components through this route.

The immediate goal of the materials program is to develop a short list of materials for longer term testing. To support the assessment of available alloys and aid in the collection of data obtained as part of the R&D program, one initial goal of the Canadian materials research program has been to assemble the large amount of available experimental data into a custom database. While still in an early stage, the database developed under this AECL-MTL joint project is proving to be a valuable tool to identify trends and key variables [11]. To date, a total of about 535 data sets on corrosion and about 35 data sets on stress corrosion cracking, spanning the time period from 1953 to 2009, have been incorporated into the database. A sample database output is shown in Figure 2.

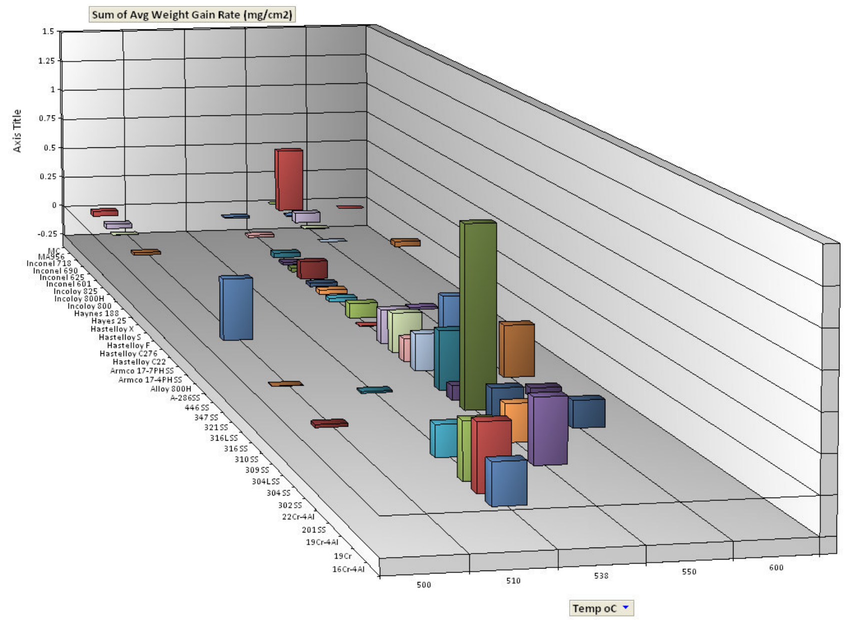


Figure 2 Variation of corrosion rate as a function of temperature, in the range of 500 to 600 ^{o}C using the data collected in the MTL-AECL corrosion database (only alloys with Cr content greater 16% are plotted) [11]

As research on the SCWR advances, gaps and limitations in the published data are being identified. With respect to corrosion properties, these gaps can be seen in several areas, including: 1) the test environment, 2) the physical and chemical severity of the tests conducted as compared with likely reactor service/operating condition, and 3) test methods used. To address these gaps, the Canadian program is continuing to develop corrosion test facilities (engineering and bench-scale SCW loops [13] and autoclaves [12], [14], corrosion fatigue testing system), high temperature (up to 850ºC) creep and creep-fatigue testers, and specialized transmission electron and focused ion-beam microscopes for handling radioactive materials samples. As part of the funding for NRCan-MTL's relocation to the McMaster Innovation Park in Hamilton Ontario, a new suite of materials processing and assessment equipment has been identified for procurement. The nuclear-materials research facilities, which are a significant part of the overall facility, can be accessed by Gen IV NSERC researcher groups through the MTL-NSERC Resource for Innovation of Engineering Materials (RIEM) mechanism.

4.2 Chemistry Research Projects

In addition to optimized material selection, the long-term viability of an SCWR requires that a set of chemistry control specifications be identified that will minimize materials degradation and radionuclide transport, optimise thermal performance and maximize system lifetime. The long-term goal of the Canadian National Gen-IV Program chemistry program is to specify suitable water chemistry (pH, dissolved oxygen and hydrogen concentrations, concentrations of any other additives, allowable concentrations of impurities, etc.) for the SCWR design. Recent work has highlighted four key areas where chemistry monitoring and control will be needed (Figure 3) [15]. A short term target is to identify candidate water chemistry specifications for long-term materials testing.

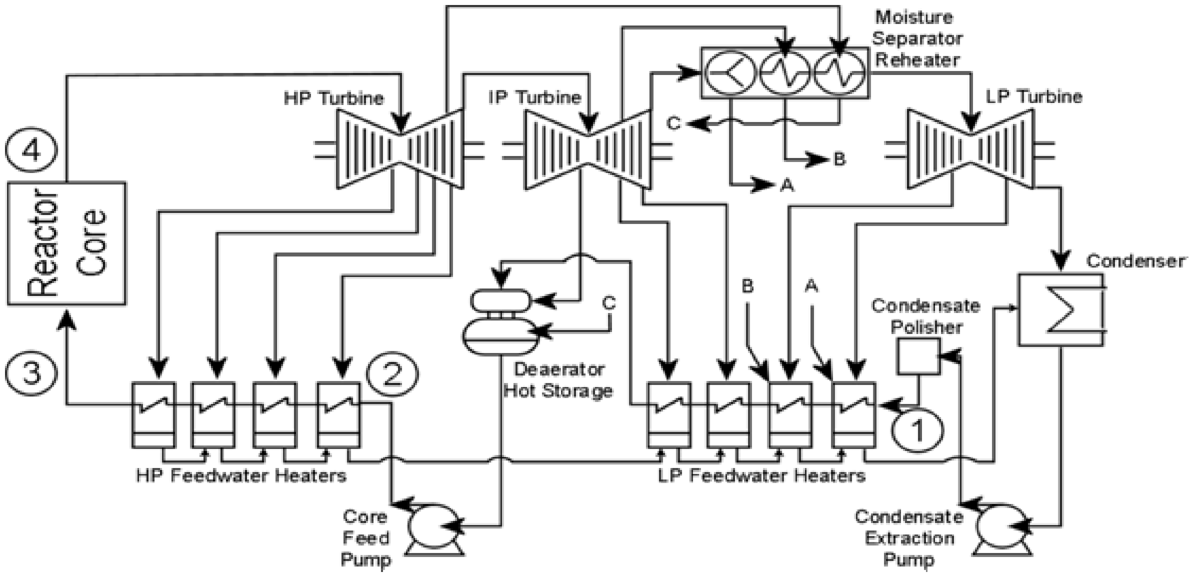


Figure 3   A hypothetical conceptual layout of an SCWR with indirect or steam reheat (for illustration purposes only). The four key locations for chemistry monitoring and control are noted.

A key challenge for the development of an SCWR water chemistry regime will be predicting and mitigating the effects of water radiolysis [16]. The radiolytic production of oxidizing species (e.g., •OH, H_2O_2, O_2, and HO_2•/O_2-•) can increase corrosion of reactor components as well as affect corrosion product transport and deposition. While current pressurized light and heavy water reactors limit the formation of oxidizing species by ensuring the presence of excess hydrogen at concentrations sufficient to chemically minimize the net production of oxidizing species by radiolysis, there are insufficient data to determine whether this strategy would be effective in an SCWR. As a consequence the coolant could be very oxidizing immediately downstream of the core (Location 4 in Figure 3).

The water radiolysis issue is being addressed by a combination of modelling and targeted experiment, with the modelling being used to identify the key gaps where experimental data are needed. A radiolysis model for the SCWR is being developed at the University of Sherbrooke; calculations of the radiolytic g-values as a function of temperature (20-400°C, 25 MPa) and water density (~0.15-0.55 g/cm^3, at 400°C) have been performed, with the focus on modelling g(e^-_{aq}), and [g(e^-_{aq}) + g(•OH) + g(H•)] [17]. While good agreement between experiment and theory is found at high “liquid-like” densities, discrepancies at low densities suggests that in this range SCW radiolysis is dominated by its “gas” phase component. The experimental program is being carried out at AECL Chalk River Laboratories (CRL), Simon Fraser University, Mount Allison University and TRIUMF. At TRIUMF, muonium (Mu) is being used as a probe of reaction kinetics in SCW; Mu kinetics below 400°C are being studied using existing equipment with aqueous samples of density from 0.25 to 1.0 g/ml, and a new high temperature cell is being developed to enable experiments to be carried out at temperatures up to 650°C [18].

The release and transport of corrosion products from surfaces of system components can result in: a) increased deposition on fuel cladding surfaces, leading to reduced heat transfer and the possibility of fuel failures, and b) increased production of radioactive species by neutron activation, ultimately increasing out-of-core radiation fields and worker dose. In addition, nuclear and thermal power stations experience deposition of steam-volatile species on turbines at levels that can cause turbine failure. Supercritical thermal station experience suggests corrosion product deposition could be significant in an SCWR [19].

While several water chemistry regimes are typically used in fossil-fired SCW plants [16], most experimental work on SCWR materials has been carried out using a limited range of water chemistries [15]. Preliminary testing under a wider range of water chemistries has been carried out at CRL to optimize the choice of water chemistries on materials corrosion. In parallel, a number of fundamental experimental and modelling studies of the aqueous chemistry of metals under SCWR coolant conditions are underway at Canadian universities, in order to obtain the data needed to develop predictive models of corrosion product transport and deposition. This work includes characterization of species present in SCW by direct spectroscopic measurements; recent work using an optimized diamond anvil cell has resulted in preliminary investigations of Mo and W oxide solubilities [20]. Thermochemical data are being measured and used to develop models for ion pairs and metal complexes in concentrated solutions, and examine the role of neutral complexes (OH-, NH3, contaminants) on transition-metal solubility in SCW. This work is coupled with quantum mechanical calculations of the stability and structure of metal complexes, estimates of solvation effects, and chemical equilibrium and mass transport modelling. The possible use of additives to minimize corrosion and/or address the effects of water radiolysis are underway, and molecular dynamics (MD) simulations of the formation of Fe(OH)2 nanoparticles in SCW have been carried out. Preliminary work [21] has provided some valuable insights. If the corrosion product loading is too high, it may be possible to remove corrosion products prior to the core (Location 3 in Figure 3), and work is underway at the University of Saskatchewan to identify and test candidate membranes for this purpose.

Finally, it will be necessary to monitor and control relevant chemistry parameters (e.g., conductivity, pH, ECP, concentrations of dissolved hydrogen and oxygen) in an SCWR and in in-reactor test loops. It is likely that reliable monitoring of key chemical parameters can only be achieved through the development of in-situ or on-line probes; this work is currently being undertaken at the University of British Columbia.

4.3 Thermal-hydraulics Research Projects

The safety design criterion for the SCWR is based on the cladding temperature limit because a phase change is not present in the core for normal operation and trip analyses. The lack of qualified experimental data on heat transfer and pressure drop for supercritical water flow has been identified as a significant risk to the SCWR design, due to the drastic deterioration of heat-transfer characteristic at the vicinity of the critical point. Fundamental understanding of thermalhydraulics characteristics has relied on experimental information obtained with tubes, annuli, and bundle subassemblies.

Experimental data obtained with tubes and bundles under supercritical water conditions are required for the development of heat-transfer correlations. Previous reviews of published information concluded that limited information is available for SCW flow in tubes [22]. These tube data provide a fundamental understanding of SC heat transfer phenomena.

Databases on supercritical heat transfer have been compiled for tubes of either upward or downward flow of water and surrogate fluids [23]. The SCW heat transfer database covers a relatively wide range of pressure and temperature. Information on the datasets has been presented in Groeneveld et al. [23].

However, the majority of the data were obtained at fluid temperatures below the pseudo-critical temperature. The database was used to assess four heat-transfer correlations: the Dittus-Boelter equation, the correlation of Krasnoshchekov et al., the Jackson correlation, and the Yang correlation [24]. Figure 4 illustrates the percentage of all vertical upflow data predicted within various error ranges.

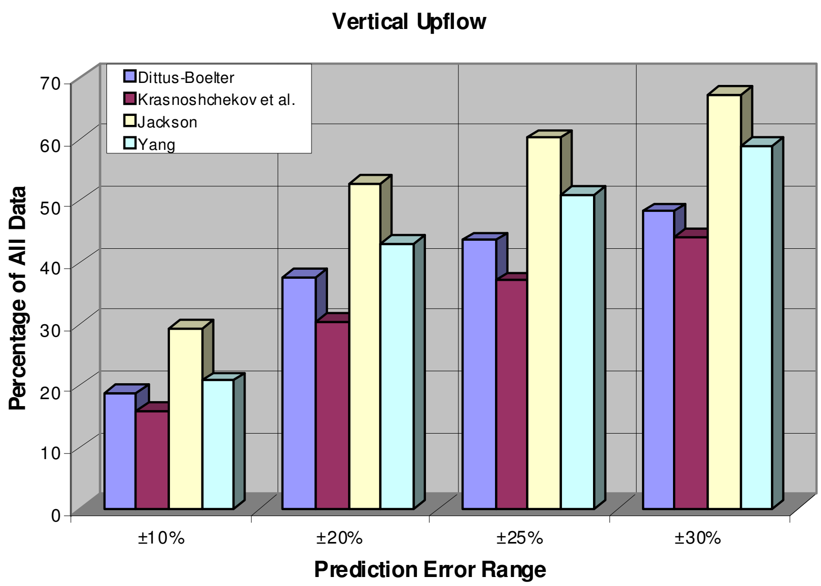


Figure 4 Percentage of All Vertical Upflow Data Predicted within Various Error Ranges

Of these four correlations, the Jackson correlation provides the best prediction accuracy for the vertical upflow database, with over 65% of all data predicted within the ±30% error range. Both the Dittus-Boelter and the Krasnoshchekov correlations predict a smaller number of data within the same error range. The prediction accuracy of the Yang correlation is comparable to that of the Jackson correlation, and is better than those of the other two correlations. These correlations tend to underpredict the wall temperature at the deterioration heat transfer region.

Figure 5 illustrates the percentage of all vertical downflow data predicted within various error ranges. The Jackson correlation again provides the best prediction accuracy. Over 85% of all data are predicted within the ±30% error range. The Yang correlation performs poorly against the vertical downflow data. Both the Dittus-Boelter and the Krasnoshchekov correlations provide similar prediction accuracy to the Jackson correlation. One of the notable differences, compared to upflow, is that the deterioration heat transfer region has not been observed in downflow. This has resulted in a reduction in scatter among data in the vicinity of the pseudo-critical point, allowing improved prediction accuracy in fuel channel heat transfer for the current CANDU SCWR conceptual design.

Supercritical heat transfer information obtained from analytical and experimental studies is being used for preliminary core design and fuel design calculations as well as safety analyses. Supercritical heat transfer correlations have been recommended from various assessments against experimental data. These correlations were implemented into safety analysis codes for preliminary safety analyses and subchannel codes for fuel design support. Due to the lack of bundle data, validation of these correlations is not feasible and hence these recommendations are considered tentative.

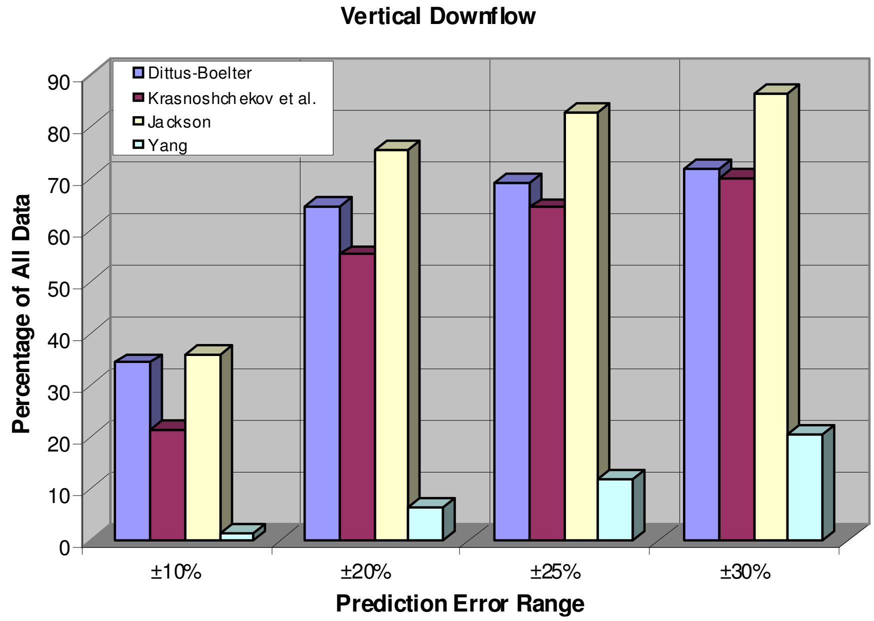
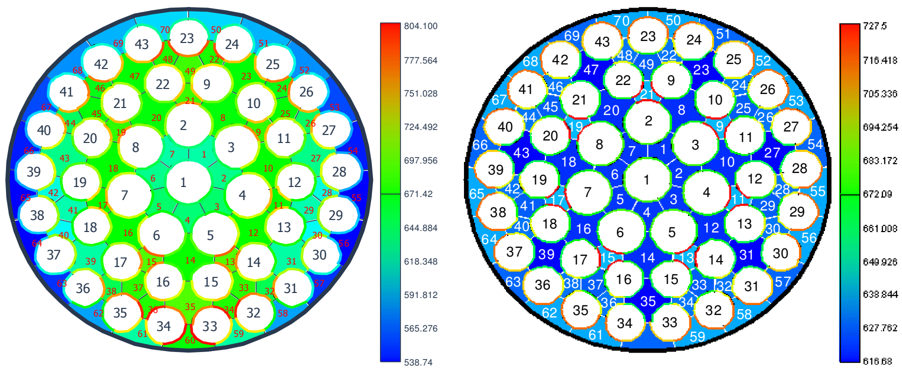


Figure 5 Percentage of All Vertical Downflow Data Predicted within Various Error Ranges

An optimization of the fuel geometry and orientation for the SCWR has been performed to lower the surface temperature (hence increasing the operating margin). The vertical channel orientation with an upflow eliminates the gravity effect on the temperature distributions. Surface temperature and coolant temperature distributions in subchannels of the CANFLEX®[2] bundle were assessed using a subchannel code based on the maximum fuel-cladding temperature criteria of 850°C for the same set of correlations for heat transfer, friction factor, and mixing. The left side of the image in Figure 6 presents distributions for the horizontal bundle of reference geometry with uniform 4% low enriched uranium (LEU) fuel, and the one on the right side shows distributions for the vertical bundle of optimized geometry with graded enrichment (from 2.5 to 6% LEU fuel). The cladding and coolant temperatures have been reduced considerably for the optimized CANFLEX bundle with graded enrichment (from 2.5 to 6%) in various element rings. The maximum surface temperature has been reduced to 727°C and the maximum coolant temperature decreases to 637°C for the vertical optimized fuel (compared to 804ºC and 693ºC, respectively, for the horizontal reference fuel). Limiting subchannels were observed at the outer ring adjacent to the pressure tube for the reference CANFLEX fuel in the horizontal channel, but have been predicted between the intermediate and inner rings for the optimized fuel in the vertical channel. This study showed improvement in heat-transfer characteristics for the vertical fuel-channel configuration in the proposed CANDU SCWR conceptual design.

[2] CANFLEX – CANDU Flexible, a registered trademark of AECL and Korea Atomic Energy Research Institute (KAERI).



Horizontal

Max. Tw: 804ºC

Max. Tb: 693ºC

Parameters	Values
Coolant inlet pressure	25 MPa
Coolant inlet temperature	350°C
Coolant exit temperature	625°C
Average coolant mass flux	890.3 $kg{\cdot}m^{-2}{\cdot}s^{-1}$
Average heat flux	0.978 $MW{\cdot}m^{-2}$
Limit of cladding surface temperature	850°C

Vertical

Max. Tw: 727ºC

Max. Tb: 637ºC

Figure 6 Cladding and Coolant Temperature Distributions in Subchannels of Reference (Uniform Enrichment) and Optimized (Graded Uranium) CANFLEX Bundles

4.4 Safety-Related Research Projects

An important metric in the design of Gen-IV reactors is improved safety. The SCWR design is considered as an extension of existing pressure-tube and pressure-vessel reactor designs. Most anticipated accident scenarios for existing reactors are applicable to the SCWR design. Therefore, safety and other systems are generally similar to those expected for modern nuclear reactors. However, some accident scenarios may be unique to the SCWR design and specific considerations are necessary to cover the transition through the critical point. To capture these unique features for the SCWR, additional safety requirements and design basis accidents must be defined for the design of reactor control and safety systems. Part of the enhanced safety features for the SCWR focuses on the design of passive safety systems.

Critical (or choked) flow phenomena are of great importance in designing reactor safety systems and analyzing the large-break loss-of coolant event. A preliminary study was performed to examine the depressurization behaviours of SCW flow. Pressure transients for the Edwards type blowdown were examined for three different fluid temperatures at an initial pressure of 23 MPa. The nodalization of the setup is illustrated inside Figure 7. As shown in the figure, the depressurization characteristics are similar to those of the Edwards type blowdown at fluid temperatures lower than the critical temperature (i.e., a rapid depressurization until the saturation point followed by a slow cool down). On the other hand, the depressurization characteristics behave similar to gas phase from a vessel at temperatures higher than the critical temperature (i.e., a rapid depressurization with no apparent levelling off at saturation conditions).

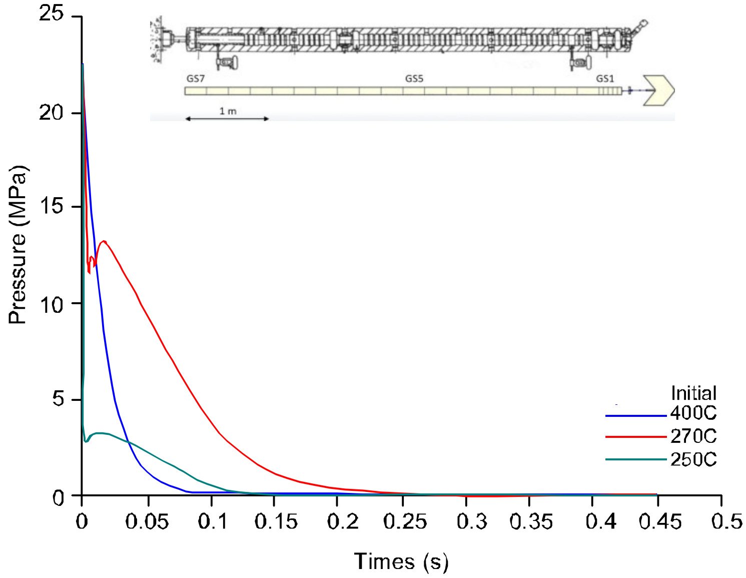


Figure 7 Pressure Variations for the Edwards Type of Blowdown Analysis

A passive moderator cooling system will be introduced to remove decay heat from the fuel in a large-break loss-of-coolant event for the CANDU SCWR. This system could potentially eliminate core melt. Decay heat from the fuel bundles is transferred to the moderator through a specially designed high-efficiency fuel channel. A passive recirculation loop is connected to the low-pressure calandria (see Figure 8). Moderator at temperatures close to saturation enters the loop near the top of the calandria. Flashing occurs inside the channel due to the change in pressure. The rise of vapour continuously drives the moderator into the loop. A secondary heat exchanger condenses the moderator vapour back to liquid, which is recirculated back to the calandria.

The concept has been proven in an experimental setup but flow instability was observed in the recirculation loop. A study has been initiated to improve the understanding of the instability mechanism and identify the instability region (or map).

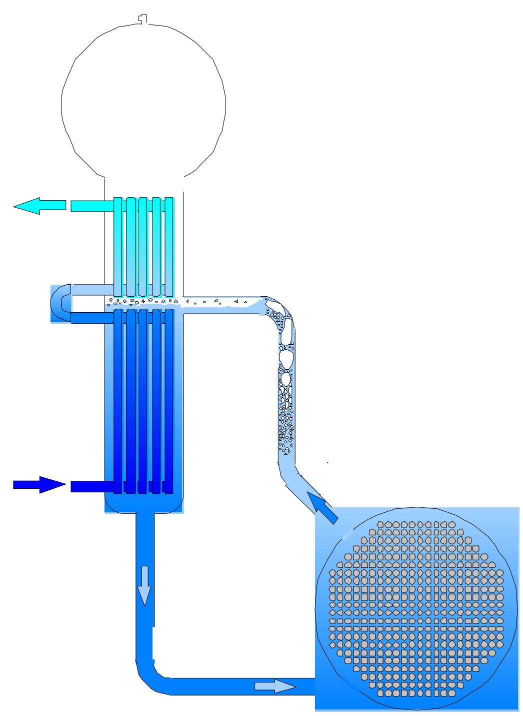


Figure 8 Schematic Diagram of the Passive Moderator Cooling System

5. Conclusion

The Canadian Gen-IV National Program has been initiated to provide R&D support to the development of the CANDU SCWR concept. Under this program, a broad range of multidisciplinary projects are being performed at AECL, federal laboratories and Canadian universities. Some of the R&D highlights from materials, chemistry, thermalhydraulics, and safety projects have been described in this paper. Until recently, the research undertaken as part of the Canadian program provided results required to guide the development of the pre-conceptual designs. Now that pre-conceptual design options have been established, the design requirements are being used to drive the R&D undertaken as part of the Canadian program.

The conceptual design option described in this paper for the CANDU SCWR maintains the characteristic modular fuel-channel configuration with heavy water moderation, but with a vertical fuel-channel orientation (rather than horizontal orientation as in the existing CANDU fleet). This transition from horizontal to vertical core configuration has many advantages, and is supported by many of the R&D results obtained to date in the program.

While preliminary results show good progress in various areas, significant gaps in knowledge and data have also been identified. To address these gaps, considerable investment is being made to build up Canada’s R&D capabilities in these areas.

6. Acknowledgement

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