Materials Challenges For The Supercritical Water-Cooled Reactor (SCWR)

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Abstract

This paper discusses the materials requirements of the Supercritical Water-cooled Reactor (SCWR) which arise from its severe expected operating conditions: (i) Outlet Temperature (to 650 C); (ii) Pressure of 25 MPa for the coolant containment, (iii) Thermochemical stress in the presence of supercritical water, and (iv) Radiative damage (up to 150 dpa for the fast spectrum variant). These operating conditions are reviewed; the phenomenology of materials in the supercritical water environment that create the materials challenges is discussed; knowledge gaps are identified, and efforts to understand material behaviour under the operating conditions expected in the SCWR are described.

1. Introduction

A number of critical drivers have combined over the last few years to generate renewed interest in nuclear energy, both within Canada and internationally, creating the momentum for what is now being called a Nuclear Renaissance. These drivers include Sustainable Energy production and use, both from strategic and environmental considerations, and for Climate Change Mitigation. Concomitantly, there is now greater motivation to replace conventional emissions-intensive industrial processes in a variety of applications, including steam generation (for example, in extraction of fossil fuels from tar sands); for hydrogen production (both for emerging transportation technologies and for refining of crude oil); as well as for desalination of both seawater and inland brackish water; among several others, by emissions-free energy technologies. Given the essential emissions-free nature of nuclear energy, these drivers have renewed the impetus to further innovate existing nuclear technology along traditional dimensions such as safety, reliability, efficiency, economics and sustainability, as well as to explore the special issues that arise in these new applications. In view of increased commercial use of nuclear technologies, there has also been stronger impetus to design reactors with constructability and modularity in mind, and to build greater proliferation resistance into fuel cycles.

These technological and strategic considerations have led to the initiation and establishment of a number of international nuclear technology cooperative development initiatives, such as the Generation IV International Forum (GIF), the International Nuclear Energy Research Initiative (I-NERI), the Euratom Project on Reactor for Process Heat and Electricity (RAPHAEL) and the IAEA-coordinated International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), among others. Canada is participating in several of these initiatives, with AECL being the primary institutional participant. Within the set of six reactor concepts proposed for R&D under the Generation-IV International Forum (GIF), Canada has taken the lead on the Supercritical Water-cooled Reactor (SCWR), recognizing (i) Considerable prior work AECL has carried out on variants of the basic CANDU[1] design using light water or supercritical water as the coolant, (ii) The fundamental complementarity between the basic CANDU design and the SCWR concept, and (iii) The facilities and expertise available in Canada to advance the R&D on the concept.

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

Canada is also participating in R&D on the Very High Temperature Reactor (VHTR), which is a graphite-moderated, helium-cooled reactor with a once-through, thermal spectrum, uranium fuel cycle, with an outlet temperature of up to 1000 C, where the United States has assumed the lead. Although VHTR is significantly different from CANDU in concept, participation in VHTR is desirable, because of (i) Likely spin-offs from its significantly accelerated development timeline relative to the other reactor concepts; (ii) The fact that VHTR development includes demonstration of process heat applications (thermochemical hydrogen production and coal gasification); (iii) VHTR development also includes demonstration of High-temperature Electrolysis (HTE). Canada has a strong interest in each of these technologies. Variants of CANDU reactors, perhaps including ACR-1000 and CANDU-SCWR, could be employed in similar applications in the future. The VHTR is more widely known as the Next Generation Nuclear Plant (NGNP) in the US, and is intended to function as such.

The new and more demanding operating conditions foreseen in Generation-IV reactor concepts arise from (i) **Fuel cycle innovations** (for example, fast spectrum neutron fluxes could result in radiation dosages up to 150 dpa on cladding materials, compared to about 15 dpa for thermal spectrum); (ii) The need for **higher outlet temperatures** for greater thermal efficiency (from about 300 C in PHWRs to about 625 C in SCWRs and 1000 C in VHTRs); (iii) **Novel coolant possibilities**, such as supercritical water, liquid sodium or molten salts, which interact more stressfully with materials comprising both the reactor core and the balance of plant than conventional light or heavy water coolants; and (iv) The **longer expected lifetime** of the reactors, at 60 years, compared to licensed lifetimes of 20-40 years for Gen-II or III reactors.

Generation-IV reactor materials will therefore be subject to higher levels of hydrostatic, thermochemical, and radiolytic stresses over a longer period than materials in currently operating reactors. Materials Issues thus form a cross-cutting R&D theme across all Generation-IV reactor concepts, and the Generation IV International Forum (GIF) has drawn up an Integrated Materials R&D Plan to address them. Many of these Materials Issues are also shared by projects whose primary goal is not reactor evolution *per se* but fuel cycle innovation or nuclear hydrogen production and, interestingly, nuclear fusion reactor development, including the International Thermonuclear Experimental Reactor (ITER).

Although materials issues are thus a cross-cutting theme, they possess a particular salience within the *SCWR* context. No *nuclear* reactor has yet been built using supercritical water as the coolant, while for the other Generation-IV concepts, even the VHTR, demonstration or experimental reactors of very closely related concepts have already been built, and coolant-material interactions are less uncertain. Significant questions exist for the SCWR, however, about how cladding materials, for example, will react to the combined radiological and thermochemical stresses in the supercritical water environment. Although many *fossil-fuel-fired* power plants using supercritical water have been built[2], there is no precedent for the type of radiological-thermochemical stress that materials in the SCWR reactor core will face, especially in a fast-spectrum fuel cycle. Though materials eventually to be used for the SCWR are expected to evolve from those currently being used in nuclear reactors generally, the issues specific to SCWR do warrant a detailed investigation. This effort is proceeding in Canada, both at AECL and at a number of universities, under the aegis of the I-NERI and the Generation-IV International Forum [1-6].

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[2]There are also other proposed industrial uses for supercritical water such as remediation of polychloro-biphenyls (PCBs) contamination, utilizing its superior oxidation properties; and supercritical-water-based chemical (non-electrolytic) hydrogen production through methanol reforming. These also generate useful data from the point of view of SCWR design and analysis.

The goal of this paper is to review these Materials Challenges with particular reference to *SCWR*, and to provide an accessible discussion of issues for the typical interested professional who is not directly working in the field. This paper is organized as follows. In the next section, a brief introduction to the SCWR reactor concept is provided, followed by a discussion of the operating conditions that create the Materials Challenges. In Section 3, the phenomenology of supercritical water is reviewed, together with what is known about materials behaviour in SCWR operating conditions. In Section 4, a discussion of the Materials Challenges is provided, including a brief discussion of the work in progress that addresses these Challenges. Section 5 is the Conclusion.

2. The Supercritical Water-cooled Reactor (SCWR)

Expressed simply, the SCWR is a nuclear reactor cooled with (light) water in its supercritical state. As is well known, when heated under sufficient pressure, water ceases to boil. The distinction between liquid and gaseous phases vanishes at pressures greater than 22.1 MPa and a temperature of 374 C. This is known as the critical point. Above this temperature and pressure, water is said to be supercritical.

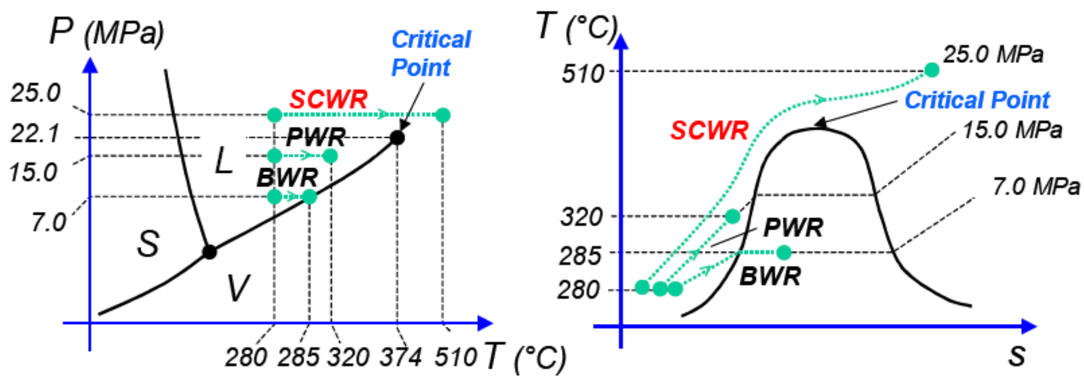

Figure 1: *Left*: Phase diagram of water, showing the critical point and the relative operating regimes of the Supercritical Water Reactor (SCWR), the Pressurized Water Reactor (PWR), and the Boiling Water Reactor (BWR). The reference temperature shown is that of the US SCWR variant, while the CANDU-SCWR outlet temperature is higher, at 650 C. *Right*: Temperature-Entropy diagram of water, showing the greater heat transfer possible above the supercritical point. From [7].

During the 1950s and 1960s, before the market dominance of PHWRs and LWRs, the SCWR concept had been proposed both in the nuclear and in the fossil-fuel-fired contexts, but interest subsequently dropped off in nuclear SCWRs. However, SCW boilers in fossil-fuel-fired power plants continued to be developed, and have now achieved very high market penetration. Even in the fossil-fuel-fired context, a number of materials issues did arise, and were significant enough to cool the initial enthusiasm for the concept. However, they were subsequently overcome, so the knowledge base generated during this process is of great relevance to the balance-of-plant design for SCWRs. Interest in *nuclear* SCWRs was eventually revived in the 1990s, through work in Canada, Japan, the US and Russia.

Much of the interest in the SCWR concept arises from the considerable expected increase in thermal efficiency (a nearly 33% increase over conventional PHWR or LWR) from the use of supercritical water as the coolant. This results from: (i) The higher range of operating temperatures that is possible; (ii) The high specific heat of supercritical water (which enables greater heat transfer per unit volume, thus permitting a lower mass flow rate compared to pressurized water as coolant); (iii) The fact that supercritical water does not change phase in the loop. These factors considerably simplify the balance-of-plant, reducing the size of pumps, piping and associated equipment, and improving the economics of the concept. Indeed, the superior expected economics of the SCWR is one of its major selling points. Use of a single-phase coolant also obviates the boiling crisis, a serious issue with PWRs, permitting temperatures to be safely raised and avoiding discontinuous heat-transfer regimes within the core, improving safety performance. Indeed, these considerations motivated the use of supercritical water as the coolant in fossil fuel fired power plants in the first place. It becomes natural, therefore, to investigate the possibility of a CANDU reactor design with a supercritical water coolant.

The synergy of the supercritical water coolant idea with the basic CANDU design is considerable. First, the physical separation of moderator (in the calandria) from coolant (in the pressure tube) reduces the coolant impact on neutron flux. Normally, this can occur from coolant and moderator both being present in high concentrations within the core. Secondly, with horizontal pressure tubes in CANDUs, the effect of density gradients within the coolant (which can be significant in supercritical water flow) as it moves through the reactor core, can be checked through bi-directional interlacing of adjacent channels, thereby balancing the density gradients by using flows in opposite directions and achieving a more axially uniform flux field [6]. This cannot be done in a vertical reactor pressure vessel (RPV) common to LWRs. In addition, the basic CANDU design can also smoothly transition to a slightly enriched uranium (SEU) fuel cycle with light water moderation, a flexibility that is a significant advantage within the SCWR design envelope. Another advantage is the relative ease of achieving a CANDU pressure tube capable of bearing a pressure of 23+ MPa that would be necessary versus achieving the same pressure in a larger Reactor Pressure Vessel (RPV) in PWRs [6].

CANDU-SCWR Reference Design Parameters

- Direct Cycle
- Supercritical Water Coolant
- Outlet Temperature 650 C
- Thermal Cycle Efficiency 45%+
- Operating Pressure 25 MPa

Figure 2: CANDU-SCWR Reference Design Parameters. (From [8]).

The CANDU-SCWR concept also has the additional virtue of being potentially scalable to a fast spectrum fuel cycle. Independently of the SCWR concept, AECL has carried out a number of studies with thorium based fast-spectrum fuel cycles [9-10]. Combining this fuel cycle innovation with the SCW coolant can lead to both higher thermal efficiency *and* a more sustainable fuel cycle, and is therefore of considerable interest as a concept for the future. The outlet temperature envisaged for CANDU-SCWR (650 C) is somewhat higher than those proposed for the US SCLWR reference design (500-550 C). The higher temperatures *may* enable some thermochemical hydrogen production, but an electrolytic hydrogen production scheme is certainly possible using the electric power produced by the reactor. However, the economics of such a system must be carefully investigated. If a secondary steam generator is set up, to provide steam for various applications, in particular, oil sands recovery, then the overall economics can improve. A CANDU-SCWR system together with an electrolytic hydrogen production process and a steam generation process has many synergies to commend itself, and can improve both the environmental footprint and the economics within the context of oil sands recovery.

Today, state-of-the-art SCW fossil-fuel plants can sustain temperatures as high as 610 C and pressures as high as 25 MPa [11]. The balance-of-plant element of the CANDU-SCWR reference design is therefore pitched at the absolute cusp of current technological possibility. As a result of the substantial continuity in the design scope of the CANDU-SCWR reactor core, however, with current generation CANDU, a considerable body of existing knowledge regarding CANDU safety issues will continue to be valid for CANDU-SCWR. Furthermore, balance of plant safety and performance is well understood from the experience with fossil-fuel SCW plants. Given all these attributes in its favour, the CANDU-SCWR concept is expected to be the mainstay of the evolutionary path of the CANDU reactor, with a development trajectory projected as far as the year 2080 [6].

A slightly different SCWR concept based on the pressure-vessel design has also been specified: (i) With a light water-moderated thermal spectrum fuel cycle (SCLWR) (ii) A fast spectrum variant, without a moderator altogether. In addition to challenges for the cladding material similar to those with CANDU-SCWR, there are significant materials challenges for the SCLWR in the materials used to build the reactor pressure vessel (RPV) as well. The general SCWR concept has also been specified within a pebble-bed context, where the fuel consists of SiC-coated UO2 ‘pebbles’, with a light water moderator. This concept has advantageous passive safety features, but some materials challenges also remain at the SCW interface. These SCWR variants are being developed in the US, Europe, and Asia.

3. Thermo-Physical Phenomenology of Supercritical Water

The thermo-physical phenomenology of supercritical water is briefly reviewed in this section, because it is pertinent to a discussion of the materials challenges in the SCWR context. The density of supercritical water is a strong function of temperature and pressure, and can easily vary by a factor of five or more in the operating temperature range of the SCWR (Figure 3).

The difference in properties between supercritical water and normal water, however, is most dramatic in the specific heat, and particularly so at the critical point itself (Figure 4). Another relevant property difference arises from the insolubility of inorganics in supercritical water, which raises the purity requirements for the intake (Figure 5).

When water becomes supercritical within the thermohydraulic loop, any inorganic impurities dissolved in it would be deposited on reactor materials. When such impurities are deposited within the reactor core, they could have harmful consequences for the thermal conductivity and structural stability of the cladding, as well as serious implications for neutronics and for reactor stability. Outside the core also, they would contribute to an increase in corrosive stress. Therefore the intake water in SCWRs should be as pure as possible; this is similar to the corresponding requirement in Boiling Water Reactors (BWRs).

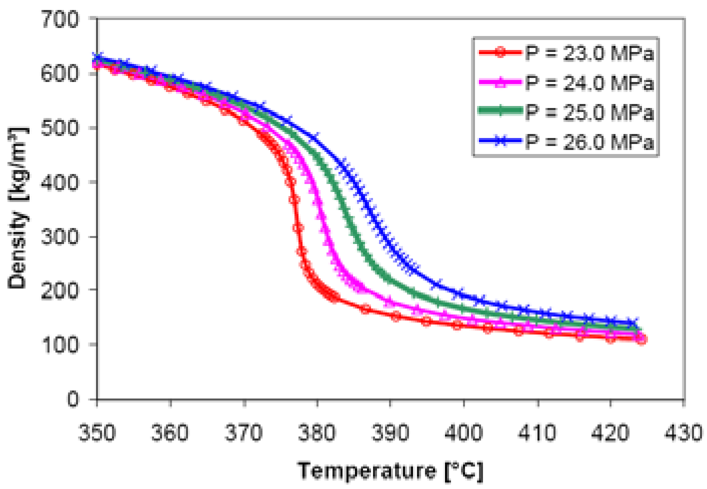

Figure 3: Density changes in supercritical water as a function of pressure and temperature. From [11]

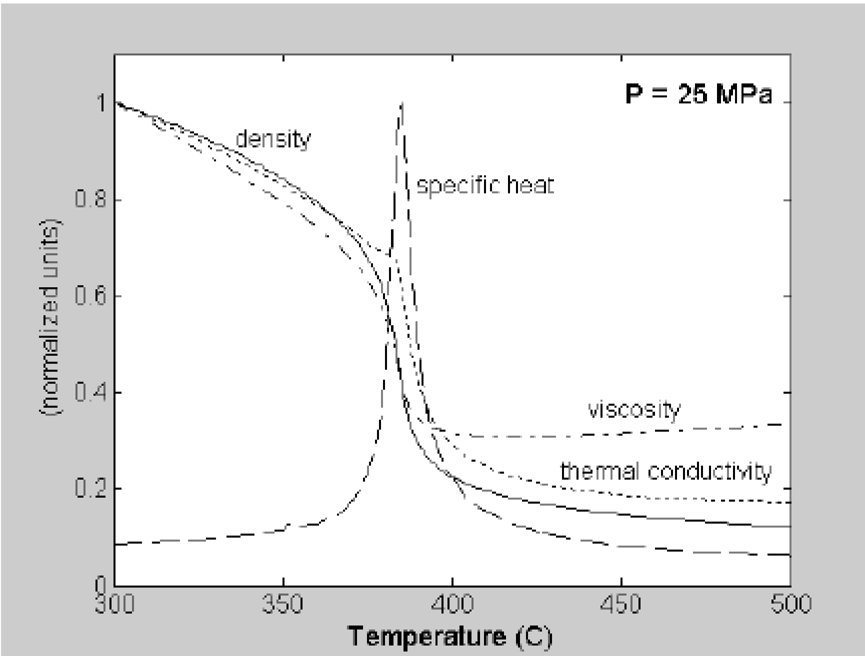

Figure 4: Thermal variation of physical properties of water, showing clearly the specific heat peak at the critical point. Density, viscosity and thermal conductivity are seen to decline significantly with temperature. A major consequence of the higher specific heat of supercritical water is that a far more efficient heat transfer can take place through the coolant, reducing the amount of mass flow required in the loop.

	Normal Light Water	Supercritical Light Water
Dielectric Constant	78	< 5
Solubility of Organics	Very Low	Fully Miscible
Solubility of Gases	Very Low	Fully Miscible
Solubility of Inorganics	Very High	Not Soluble

Figure 5: A comparison of some relevant properties of normal light water and supercritical light water.

4. Materials challenges

The feasibility of the SCWR concept will be decided based on whether materials can be found that can withstand the combined thermal, hydrostatic, thermochemical and radiative stresses arising from the operating conditions over the lifetime of the reactor. This is a particular issue for in-core reactor materials, and less so for balance-of-plant materials, which can use materials previously tested in fossil-fired SCW plants. A brief review of the stainless steels and alloys that are currently used in nuclear plant structures is first presented, along with some basic water chemistry. The discussion will then cover effects of high temperature, pressure, and radiation; candidate materials for SCWR will then be outlined. Finally, a brief description of the ongoing experiments and plans is provided. A more comprehensive discussion is available in [11].

4.1 Ferritic, Austenitic and Martensitic Steels

Steels are classified by their structure into *ferritic*, *austenitic* and *martensitic* steels. *Ferritic* steels are highly corrosion resistant, but not as durable as austenitic steels. *Austenitic* steels are the most widely used stainless steels; over 70% of stainless production is austenitic. They are characterized by a face-centred cubic microstructure. Austenite is stable only above 723°C in carbon steel, but alloying elements such as nickel and manganese, stabilize it at low temperatures. *Martensitic* steels are named for the distinctive needle-like structure (martensite), which comes into being as austenitic steel transforms during quenching. Currently, martensitic and ferritic steels and their nickel, manganese, carbon and chromium alloys are widely used in reactor materials.

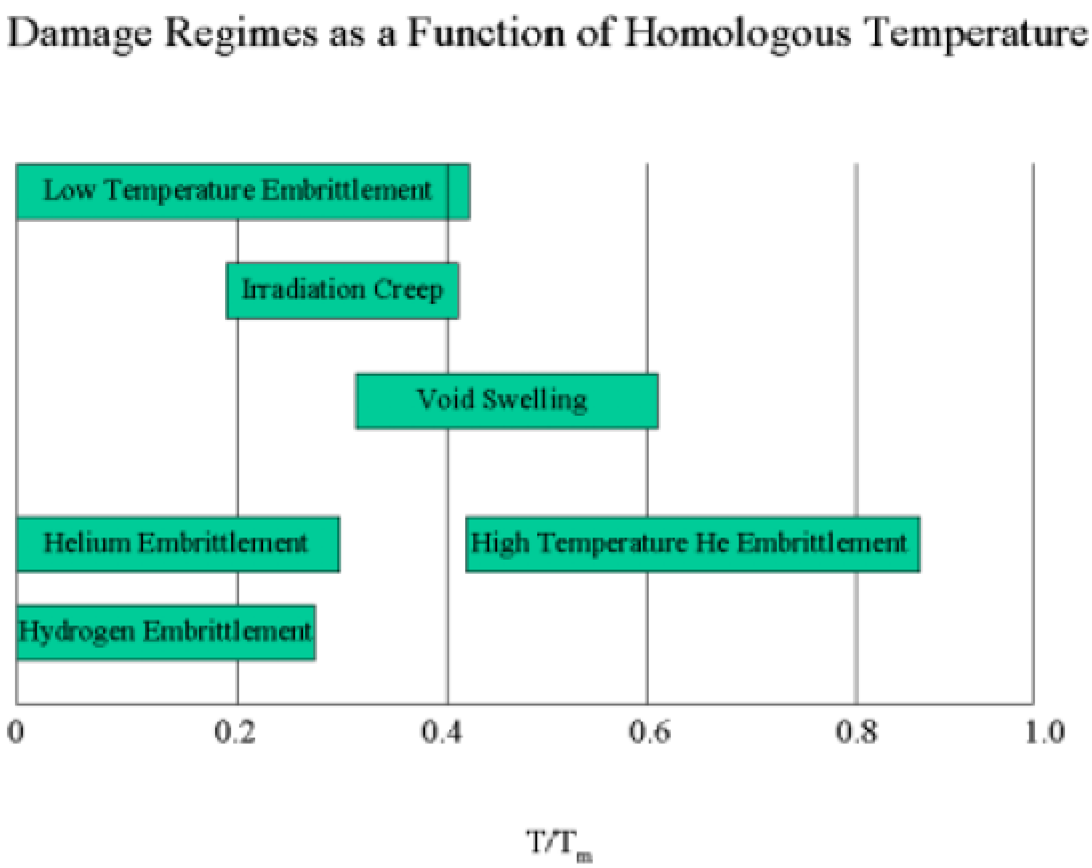

Figure 6: A diagram sketching out the different damage regimes applicable to alloys used in reactor materials, as a function of the homologous temperature. (Homologous temperature is the temperature in fractions of melting temperature.) From [12].

Reactor materials are stressed by four main factors (and their collective impact): (i) High temperatures. (ii) High Pressures (iii) Thermochemical environment (iv) Irradiation flux.

4.1 High Temperatures and Pressures

The high operating temperature and pressure in SCWRs can induce a heavy stress on the structures. There is usually a maximum pressure that structures can handle at a given temperature without failure, defined by the bulk modulus of the structural material. But interestingly, this pressure threshold rises with temperature, and most structures in the power plant are designed to handle the operating pressure only at higher temperatures. Even though structures can be designed in this way to withstand stresses without breaking, long-term phenomena like *creep* pose significant challenges. Creep is the slow plastic deformation of materials under constant stress. Creep occurs when the large number of pre-existing lattice dislocations in a given material start moving slowly in response to the external load. At the microstructural level, these dislocations reach a metastable equilibrium, resulting in the hardening of the material. At low temperatures, the material would freeze in this condition, but at higher temperatures, the vibration of atoms creates restoring forces, which tend to repair the dislocations. Cyclic variations in the pressure can cause additional stresses to structures. These stresses appear at lower values than the critical steady pressure for a given system, and must be properly considered during the plant operation. The limiting factor for the operating lifetime of the reactor is creep that accumulates over the years. Also, the combined effects of chemical corrosion and stress on the materials may cause *stress corrosion cracking* (SCC).

4.2 Irradiation

When a material is irradiated, all of its physical properties can change. Physical dimensions as well as the electrical, mechanical, magnetic, thermo-physical and other properties can each change. During irradiation, the kinetic energy dismantles the atomic lattice first, and lattice-restoring forces then reconstruct it, atom-by-atom. In high-dose irradiation, each atom may be displaced from its lattice site many times, the standard measure of which is the displacements per atom (dpa). The specific conditions at the time of irradiation, such as temperatures, and local material composition, in addition to dose and dose rate, ultimately determine the property changes that will result. The irradiation-induced changes of greatest concern are (i) Swelling, (ii) Irradiation creep, and (iii) Embrittlement.

4.2.1 Swelling

Swelling is the isotropic volume expansion of an irradiated material, occurring by the net absorption of interstitials at dislocations, with a corresponding net number of vacancies accumulating at cavities. At high doses, tens to hundreds of dpa, swelling may reach several tens of percent or more of original volume. In graphite, which has a very anisotropic crystal structure, swelling can itself be anisotropic and is highly dependent upon the graphitic microstructure and the macroscopic direction of a component with respect to the crystal texture.

4.2.2 Irradiation Creep

Irradiation creep is the slow change in the shape of a material in response to an applied radiative stress, over and above the ordinary thermal creep described above. Irradiation creep can occur even at low temperatures, where thermal creep is largely absent.

4.2.3 Embrittlement

Embrittlement takes place through two processes. In the first, the hardening results from *microstructural movement of the dislocations*, which reduces ductility. The second type of process is *grain boundary weakening*, caused by preferential diffusion of transmutation products, such as helium, or tramp elements, such as phosphorus, to the grain boundary. Either process has the effect of reducing the elasticity of the material and lowers the pressure threshold.

4.3 Water Chemistry

The single most important variable that is likely to impact the practical operation of the SCWR is the chemistry of supercritical water in the presence of radiation. While the effects of water chemistry will be most critical in the SCWR reactor core, there could also be 'spillover' effects on the balance of plant systems. Control of the chemical composition of the coolant water is therefore very important. The observed mechanisms for chemical environment-sensitive cracking in water-cooled reactors are (i) Intergranular stress corrosion cracking (IGSCC), (ii) Irradiation-assisted stress corrosion cracking (IASCC) and (iii) Corrosion fatigue. These phenomena are affected by:

- Metallurgical structure, phase morphology, and depletion of metallic species such as Chromium from zones adjacent to grain boundaries;
- Irradiation effects on grain boundary impurity segregation; and
- The presence of oxidizers and reducers dissolved in the water.

IASCC in austenitic stainless steels is more significant above a *radiative fluence threshold* of about 1 displacement per atom (dpa). Further, in nickel-based super alloys IASCC is sensitive to the presence of impurities such as phosphorous, silicon, boron, or sulphur. The question of how structural materials previously used in PWRs or LWRs will perform in SCWRs is uncertain. The details will depend on precisely how the SCW water chemistry is different. Specifying the operating temperatures in the SCWR does not by itself automatically determine the water chemistry, the major reason for uncertainty. The concentrations of the transient and stable species which are formed through (i) radiolysis of the water in the presence of radiative flux, and (ii) thermal decomposition of the water due to the higher operating temperature, may well be significantly different from the situation in LWRs and PWRs. The situation will be exacerbated in a fast spectrum fuel cycle environment, where a higher radiative flux will likely radiolyse more water, changing the equilibrium concentrations of the transient and stable species. The chemical potentials of oxygen and hydrogen peroxide that are formed in this process also affect the corrosion potential of the water. It can be expected that these potentials will be significantly different in supercritical water as compared to subcritical water. The concentration of these species also determines whether magnetite (Fe3O4) or hematite (Fe2O3) will form during oxidation, and it will also affect the actual morphology of the oxide films, which are important to corrosion control in steels.

Similarly, the chemical potential of the hydrogen is likely to be different in supercritical water, just as the chemical potential of the oxygen would be, and the water chemistry of hydrogen could be effective in reducing the oxygen content. The reaction rate of the OH radical with hydrogen has been known to decrease above 300 C, increasing the relative probability of thermolysis of the water molecule and the equilibrium concentrations of the ionic species. In the reactor core, water will thus be radiolysed, but since this process is kinetically determined, it might require much more hydrogen to suppress the oxygen and peroxide generation. If too much hydrogen is required for oxygen suppression, metal hydrides could form. The trade-off between (i) hydride formation and (ii) oxygen and peroxide generation will mainly determine how much of the LWR and fossil plant water chemistry control experience is applicable to the SCWR. The pH of the water is important in setting the corrosion potential and rate, and to some extent, also the mode of corrosion. A range of pH has been previously successfully employed in PHWRs and LWRs, and the applicability of this approach to SCWRs will need to be explored. The control of pH, while theoretically possible, may be difficult in practice, however, especially in the 300 to 650 C temperature range [13].

While the SCWR itself is unprecedented, there does exist a body of experience regarding performance of reactor materials in water environments developed in the operation of LWRs and supercritical fossil-fired power plants that may be relevant – either directly, or after appropriate interpolation. Control of the water chemistry will be critical to the continued operation of the SCWRs just as it has been for LWRs.

Boiling water reactors (BWRs) also normally operate with an oxygen overpressure. The water environment in BWRs tends to be slightly acidic because of the CO2 in air, which leads to formation of carbonic acid. The consequence is another rather aggressive environment, which, though qualitatively different from that expected in SCWRs, can still be expected to cause excessive corrosion of the reactor materials. In BWRs, the rough rule of thumb is that the propensity for SCC will increase with increasing oxygen content, so hydrogen is added to the intake water to recombine with the thermolysis-generated oxygen, and thus limit the corrosion to a value below the threshold for onset of SCC. However, significant hydrogen overpressure is needed to induce recombination of oxygen with hydrogen. More recently, thin layers of noble metals (e.g., platinum and rhodium) have been deposited on the surface of BWR structural materials to suppress the corrosion potential, and then a relatively low hydrogen injection level is sufficient.

The PWR on the other hand, is less susceptible to air infiltration, operating an indirect cycle. However, even PWRs have an oxygen overpressure due to diffusion of radiolytic hydrogen out of the coolant system. Hydrogen is therefore also injected in the primary coolant of PWRs, but at lower rates than in BWRs. A minimum temperature of about 300 C and a pH of 6.9 is required to avoid heavy crud deposits on the fuel rods. Boron is also added to the PWR coolant as boric acid to act as a neutron absorber for reactivity control. To counter the effect of the boric acid on the pH, lithium hydroxide is dissolved into the PWR primary water. In once-through fossil-fired units of the type considered for SCWRs, pre-treating the intake water usually controls the quality of the coolant water. Some combination of these and similar strategies, which must be studied in detail for SCWRs, will be needed to address the water chemistry – radiative stress environment for supercritical water and the challenge it represents to material structural integrity [13].

4.4 Materials Selections and R&D for SCWR

Given that the three factors that will most affect the properties and choice of the structural materials from which the SCWR components will be fabricated are (i) effects of irradiation, (ii) high-temperature exposure, and (iii) interactions with both the sub- and supercritical water environment; an extensive testing and evaluation program is required to assess the effects that these factors have on the properties of potential SCWR materials. The overall goal is to enable a preliminary selection of the most promising materials to be made, and to then qualify those selected for the service conditions required. This effort is already underway, and among several ongoing projects in Canada, [4] provides an overview of the Materials Evaluation for SCWRs under I-NERI at AECL.

That project is designed to investigate issues such as mechanical properties, dimensional and radiation stability, corrosion resistance, and stress corrosion cracking (SCC) in the candidate materials. While details are available [4], only some general considerations will be discussed here.

The greatest materials challenge presented by the SCWR will be in the qualification of materials that experience *both* high temperature and radiation exposure and which must simultaneously survive the hostile supercritical water environment. The fuel cladding is perhaps the material component that will receive the highest levels of this type of combined stress. Also, since modified pressure tube designs with an insulator are being considered specifically for SCWRs, their materials will have similar requirements [6]. While fuel cladding will be periodically replaced as reactors are refuelled, other structural components should have a much longer expected lifetime, so both types of components will have demanding requirements. Fast spectrum fuel cycles will offer even more stringent demands on the materials.

In structures where the temperatures will be significantly above 300 C, or radiation doses above several dpa are expected, as is true for the SCWR reactor core, the candidate structural materials are primarily Ferritic or Martensitic steels and low swelling variants of Austenitic stainless steels. The range of compositions within the Fe-Cr-Ni alloy system with acceptable mechanical behavior and dimensional stability that currently exist, or could be developed, are in four broad categories: a) austenitic stainless steels, b) F/M steels, c) high alloys (Fe <50 wt.%) and d) Ni-based alloys. However, currently there is not a sufficient knowledge base for predicting the stress corrosion cracking (SCC) or irradiation-assisted stress corrosion cracking (IASCC) behavior under supercritical water conditions.

Some alloys have also demonstrated low swelling in doses of up to 50-100 dpa in both mixed spectrum and fast reactors in the temperature regime of 450-550 C. Ferritic-martensitic steels in the 9-12% Cr range are intrinsically more swelling resistant than austenitic steels. Low swelling has been demonstrated in these alloys at doses of 50-100 dpa in neutron irradiations. In recent years, a class of advanced ferritic steels has been the focus of strong interest for nuclear applications: the Oxide-Dispersed-Strengthened (ODS) steels. In the ODS steels, the cubic-centered structure provides the irradiation swelling resistance while the dispersed oxides (e.g., yttrium oxides) provide enhanced high-temperature strength. The high-temperature creep strength of these alloys is exceptional even at 650 C (the reference outlet temperature of the CANDU-SCWR [8]). Significant international activities are ongoing to develop and optimize this class of materials, and work in Canada is also investigating them [4].

However, the principal issues with all ODS alloys relative to their application in the SCWR remain (i) significant uncertainties regarding their interaction with the supercritical-water coolant, (ii) high cost of fabrication and (iii) weldability. Nevertheless, because of their potential, ODS alloys are being investigated carefully in the SCWR materials R&D program.

5. Conclusion

The Supercritical Water-cooled Reactor, SCWR, one of the most promising of Generation-IV Reactor concepts, with its (i) many synergies with the basic CANDU design; (ii) superior economics; (iii) higher thermodynamic efficiency; (iv) potential for advanced fuel cycles (SEU or thorium-based); and (v) diverse application possibilities in hydrogen production, coal gasification and electrolysis – has been adopted as the mainstay of the likely evolutionary trajectory of the CANDU reactor, known in this context as CANDU-SCWR [6].

Before the CANDU-SCWR design can be fully specified, finalized and operationalized, however, significant challenges must be addressed (*a*) in understanding material behaviour in the operating conditions that are foreseen, and then (*b*) in down-selecting and qualifying suitable materials for use in its reactor components. The effort to address these materials challenges is well underway in Canada [4]. Projects to support these efforts are being carried out (i) at AECL; (ii) at universities; (iii) under university-AECL collaborations; and (iv) under international cooperative initiatives. This effort will remain active well into the next decade [8], as part of a proposed development timeline that could see an operational CANDU-SCWR during 2025-2060 [6].

This paper has provided a general discussion of some of the most important challenges facing materials to be used in constructing the Supercritical Water-cooled Reactor, including (i) basic thermo-physical properties and (ii) radiological phenomenology of (*a*) candidate reactor materials and (*b*) supercritical water. General considerations and R&D issues regarding some candidate materials including ODS steels, have also been outlined. It is the author's hope that the paper will serve as an accessible overview of the field for the interested professional not directly working in it.

8. References

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