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MATERIALS REQUIREMENTS FOR THE CANADIAN SCWR CONCEPT

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Materials Requirements for the Canadian SCWR Concept

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Abstract

The high operating pressure and temperature of the Canadian SCWR raise the material-selection requirements for various in-core and out-of-core components. Material selection for the feedtrain is based on boiling water reactor and SCW fossil-power plant best practices, while selection of components downstream of the core is based on the latest advances in material development for ultra-supercritical fossil-power plants. The most challenging requirements are for in-core components. While high nickel stainless steels or nickel-based alloys are considered appropriate for components outside the irradiated region (e.g., inlet and outlet plenums, tubesheet, etc.), these materials may not be appropriate for components in regions (e.g., fuel-assembly) subjected to a high neutron flux due to the unfavorable neutron absorption characteristics of nickel. A number of stainless steels appear suitable for in-core components that encounter a maximum coolant temperature of 625 °C. However, these materials may not be suitable for use as the fuel cladding, as the current maximum design limit for the fuel-cladding temperature is 800 °C. A significant effort has been devoted to assessing potential candidate fuel cladding alloys.

A collapsible cladding has been selected for the Canadian SCWR fuel, reducing the material strength requirements. Corrosion, cracking, creep and irradiation damage then become the limiting phenomena. Corrosion leads to thinning of the cladding wall, challenging its integrity at high burnup when the internal pressure of the fuel element due to fission gas build-up becomes high. Furthermore, the build-up of corrosion products on the cladding surface will reduce the effectiveness of heat transfer from the cladding to the coolant, leading to high cladding and fuel temperatures. Cladding cracking on both the coolant and fuel sides can lead to fuel failure and must be assessed. This paper outlines the key materials requirements, major knowledge gaps, and the program in place to address these gaps.

1. Introduction

Canada is developing a Super-Critical Water-cooled Reactor (SCWR) concept with operating conditions compatible with the advanced high-pressure turbines designed for SCW fossil power plants (i.e., turbine-inlet pressure of 25 MPa and temperature of 625 °C). Evolving from the well-established CANDU reactor, the Canadian SCWR concept is based on a pressure-tube configuration [1] and maintains key features such as separating coolant and moderator, and low-pressure heavy-water moderator. The reactor core contains 336 fuel channels producing 2500 MW(t) (1200 MW(e) at 48% thermal efficiency), each housing a 5-m fuel assembly consisting of 62 elements filled with pellets of mixed thorium and plutonium (13%) fuel. Unlike the CANDU fuel channel, the Canadian SCWR fuel channel has only one open end. From the inlet plenum the coolant travels down a central flow tube inside the fuel assembly to the bottom of the fuel channel at a relatively constant, sub-critical temperature, then travels upward through the fuel elements where it is heated to 625 °C and discharges into the outlet plenum, which is connected directly to the high-pressure turbine.

1.1 Materials Selection Considerations

The coolant system of a typical SCWR concept (Figure 1) consists of: 1) the feedtrain, which takes condensate from the turbines, purifies it to remove impurities, and then reheats it to the core inlet temperature through a series of low- and high-pressure heaters; 2) the reactor core, which heats the feedwater through the pseudo-critical point by passing it over the nuclear fuel; and 3 and 4) the main steam line, turbines and generator, which convert the thermal energy of the coolant into electrical energy.

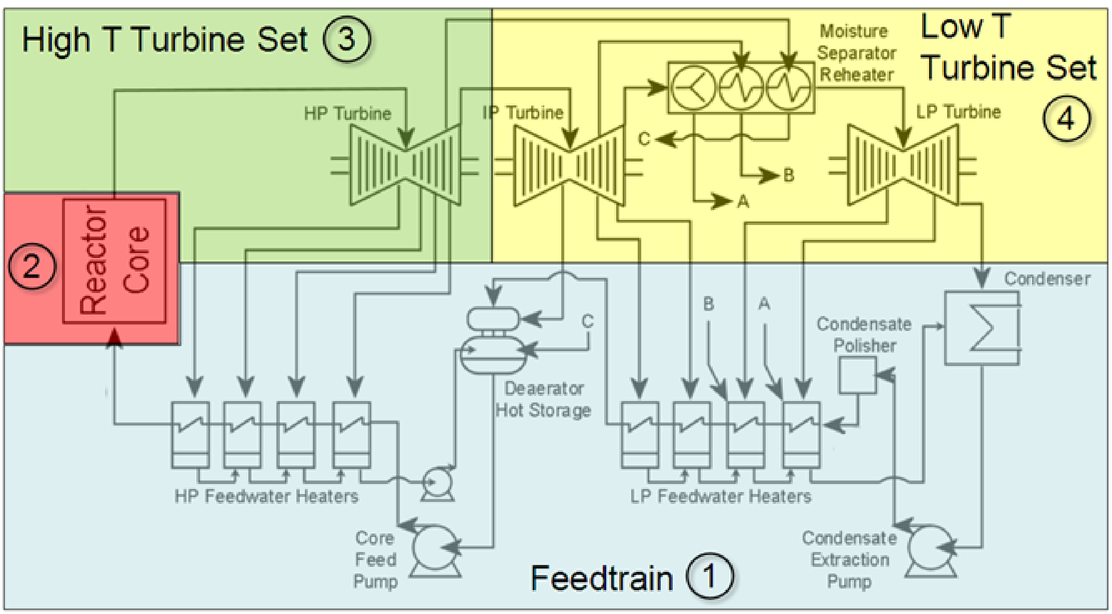

Figure 1: A conceptual layout of a direct-cycle SCWR with steam reheat. The circuit has been divided into four regions (shaded): Feedtrain, Reactor Core, High Temperature Turbine set, Low Temperature Turbine set.

It is important to note that only a portion of the core piping, the main steam line and the high pressure turbines (regions 2 and 3 in Figure 1) are at supercritical temperatures. The rest of the system operates at temperatures for which a significant amount of operating experience and a well-developed knowledge base exist. As a result, material selection for the feedtrain will be based on best practices from the boiling water reactor (BWR) and SCW fossil-power plant industries, while selection of components downstream of the core will be based on the latest advances in material development for ultra-supercritical fossil-power plants (USC FPPs) [2]. Materials development for USC FPPs is an active area of research, much of which is synergistic to SCWR development. Feedtrain and downstream material selection will not be discussed further in this paper.

Obviously, the key difference between an SCWR and an USC FPP is the various effects of radiation. The need for neutron economy dictates the use of a thin fuel cladding (relative to the thick boiler tubes in a fossil-fired plant), which places very stringent requirements on cladding integrity in order to avoid large fuel defects. While small defects may be acceptable (although undesirable), large defects will rapidly contaminate the system making maintenance difficult or impossible and increasing operating costs. Release of radioactive material into the environment outside of plant containment is socially unacceptable. In addition, irradiation of the coolant and in-core materials leads to various forms of degradation that are not encountered in fossil-fired plants.

Irradiation of materials leads to a number of different forms of damage. Irradiation damage is often quantified in terms of displacements per atom (dpa), the number of times an atom is displaced from its normal lattice site by atomic collision processes. The effects of irradiation on various alloy types have been recently reviewed [3, 4]. The most important observable physical changes in material properties resulting from irradiation are embrittlement, radiation induced growth and swelling, creep, and phase transitions. Embrittlement can seriously affect the performance of a component. Radiation induced growth and swelling can change the geometry of a component, affecting processes such as coolant flow and control rod movements. Irradiation creep is a permanent deformation caused by the evolution of irradiation-induced defects; the material grows in a particular direction and does not return to its original dimensions when the applied stress is removed. Irradiation can result in phase transitions leading to negative (and sometimes positive) changes in radiation resistance. Helium and hydrogen atoms produced by (n, γ) and (n, p) reactions can coalesce into gas bubbles that can produce voids and swelling. Neutron-induced transmutations can significantly alter the elemental composition of materials and adversely affect their properties. The transmutations also lead to the production of radioactive isotopes in the materials. If these isotopes are released into the coolant by corrosion or mechanical processes (wear, oxide spalling), they can be transported out of the core and deposit on out-of-core surfaces, leading to high radiation fields (activity transport). Guzonas and Qiu [5] have recently outlined the key aspects of activity transport in an SCWR. To minimize the effects of activity transport, certain elements must not be used in materials to be placed in the reactor core. In particular, cobalt must not be used as irradiation of naturally-occurring ^{59}Co produces ^{60}Co, which has a 5.2 y half-life for radioactive decay and emits two energetic (hence penetrating) gamma rays.

As all materials interact with neutrons to varying degrees an appropriate balance must be struck between reactor physics and material selection. Pencer et al. [6] performed lattice physics calculations for several categories of candidate in-core materials using the Canadian SCWR core design, and derived a simple relation that could be used to estimate the relative influence of in-core materials on lattice reactivity and fuel discharge burnup from material composition and density. Changes in exit burnup relative to a reference were shown to be proportional to changes in the macroscopic thermal neutron absorption cross section, allowing a qualitative ranking of materials with respect to their impact on fuel efficiency based on their thermal neutron absorption cross sections. This qualitative ranking was shown to be independent of the details of the channel, bundle or lattice.

1.1.1 Water Chemistry Considerations

SCWR water chemistry, discussed in detail elsewhere [7, 8], is closely linked to materials performance in SCW, as shown schematically in Figure 2.

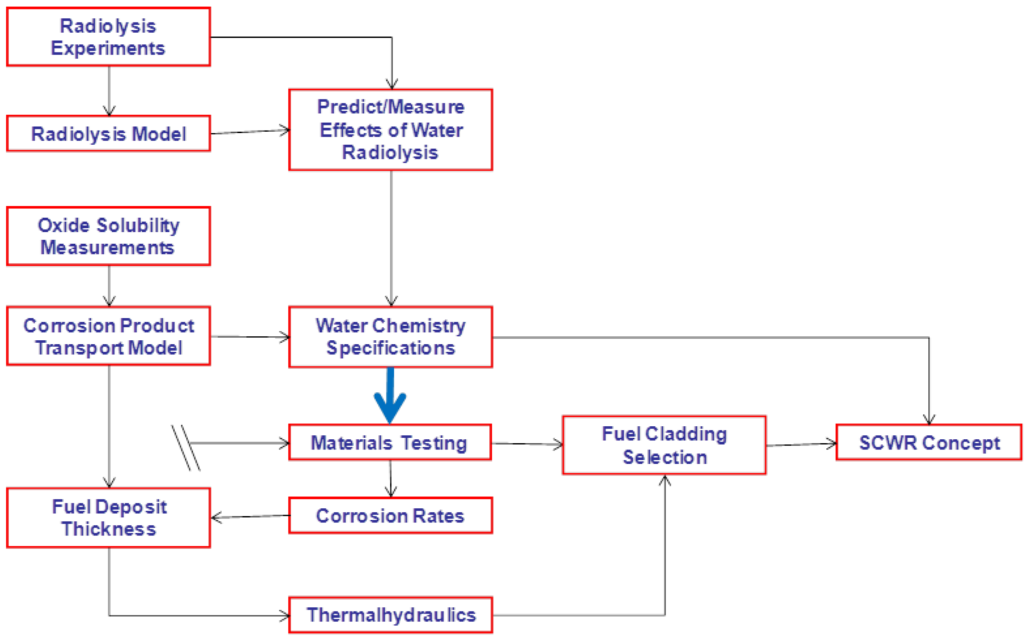

Figure 2: Link between water chemistry and fuel cladding selection.

While water chemistry will not be discussed in detail in this paper, two key chemistry issues must be highlighted in relation to materials selection. The first is the importance of impurities such as chloride on the degradation of candidate SCWR materials [9]; testing during the US nuclear reheat development program in the 1960s found that chloride deposition eventually led to failure by stress corrosion cracking (SCC), even with the best available efforts to remove chloride. An early study of chloride deposition in the BONUS[1] reactor showed that wet steam containing chlorides and oxygen caused chloride-induced SCC failure of Type 304 and Type 347 stainless steels. Experiments by Bevilacqua and Brown [10] showed that chloride deposition from the drying of moist steam resulted in heavy, adherent localized deposits. These heavy deposits, together with the presence of oxygen and water, were conducive to severe chloride-induced SCC of austenitic steels. It should be noted that Unit 2 at the Beloyarsk Nuclear Power Plant (a pressure-tube boiling water reactor with nuclear steam reheat channels) operated for many years with a typical chloride concentration of 25 $\mu g \cdot kg^{-1}$ with no reported negative effects [11]. However, tests carried out in support of the Beloyarsk NPP showed that the stainless steel used for the channel elements (1Kh18N10T) cracked after 144-1100 h of temperature and pressure cycling in an environment containing chloride, due to stress corrosion cracking. It was suggested that deposition of moisture on the outer surface and subsequent evaporation may have led to chloride concentration on the surface [12]. No testing with representative concentrations of chloride have been carried out in support of the various SCWR concepts; current BWRs operate with feedwater chloride concentrations as low as 0.25 $\mu g \cdot kg^{-1}$ [13].

[1] BOiling NUclear Superheater, a nuclear steam reheat test reactor developed in the United States. It started operation in 1964.

The second issue is that the SCWR coolant will be subjected to an intense radiation field as it passes through the reactor core. Water radiolysis reactions resulting from the ionization of water by the passage of γ- and fast-neutron radiation (and β-radiolysis close to fuel cladding surfaces and fission fragment radiolysis from tramp uranium) leads to the formation of hydrogen, oxygen and hydrogen peroxide. Recent simulations of Yeh et al. [14], although subject to significant uncertainties, showed that very high concentrations of oxidants are possible in an SCWR core if water radiolysis is not controlled.

Of particular concern for SCWR materials selection is the behaviour of Cr under oxidizing conditions; the passive films formed on almost all alloys being considered for in-core use in an SCWR are Cr-oxides. Chromium oxides are soluble under oxidizing conditions [15] due to the formation of soluble Cr(VI) species. Chromium oxide dissolution as H_2CrO_4 (transpassive dissolution) is noted in BWR plants that operate with normal water chemistry (i.e., no hydrogen addition) because of the high concentration of oxidizing species present from water radiolysis. Recent measurements [16] using an SCW convection loop with an irradiation cell coupled to a 10 MeV, 10 kW linear electron accelerator [17] clearly demonstrate the risk of Cr oxide dissolution in an SCWR core. In ~500 h duration tests of candidate alloys at temperatures in the vicinity of the critical point under electron irradiation, the outlet sample water was found to contain 3-5 $\mu g \cdot kg^{-1}$ of oxygen; due to the long sample line the actual oxygen concentration in the test sections was likely higher. The conductivity of the water at the test section outlet increased systematically up to ~23 $\mu S \cdot cm^{-1}$ indicating release of metals into the coolant due to corrosion. Elemental analysis of the water indicated the presence of Cr at concentrations up to 54 $\mu g \cdot L^{-1}$; there was no detectable Cr in the water before exposure to the electron beam. It is important to note that most of the materials testing in support of the development of nuclear steam reheat in the US was carried out with concentrations of dissolved oxygen in the range of 20-50 $mg \cdot kg^{-1}$.

1.1.2 Candidate Alloys

Neutron economy is a major factor in the design of a nuclear reactor. Zirconium (Zr), which has a low neutron capture cross section, remains the preferred metal for fabrication of reactor core components. However, at temperatures above about 450 °C a large increase in the corrosion of Zircaloy-2 and Zircaloy-4 has been reported [18–20]. Therefore, conventional Zr alloys do not appear to be acceptable for use as a fuel cladding material in an SCWR.

The prime candidate materials for in-core use in the various SCWR concepts are austenitic stainless steels. For example, the Japanese have proposed a Zr-modified 310 stainless steel as their primary fuel cladding candidate [21]. The EU have examined stainless steels such as 1.4970, 347H or 316, although they note that the corrosion resistance of these materials would need to be improved without compromising the creep and fatigue strength [22].

Some high-nickel alloys possess a unique combination of physical and mechanical properties that allow them to serve extended in-reactor life-times over a wide range of temperatures. Unlike austenitic or ferritic/martensitic steels, many high-nickel alloys also have greater high temperature strength. These alloys often have a higher resistance to void swelling and thereby a lower irradiation creep rate. The effects of irradiation on nickel-based alloys was studied during fast reactor development programs, largely because of their high resistance to radiation-induced void swelling compared to austenitic steels, although there were concerns regarding their susceptibility to irradiation embrittlement. The Nimonic alloy PE16 was successfully used for fuel element cladding and subassembly wrappers in the United Kingdom, and Inconel 706 was utilized for cladding in France [4], and its mechanical properties have been studied after irradiation to relatively high dpa values at temperatures up to 650 °C [22].

The next sections will discuss core materials selection, with a focus on identifying critical path items.

2. Materials Selection

Figure 3 illustrates the Canadian SCWR core concept [1], and serves to highlight the key components, including the calandria, fuel channels, coolant cross-over point, and inlet and outlet plena. Materials selection for each of these will be discussed in the next sub-sections.

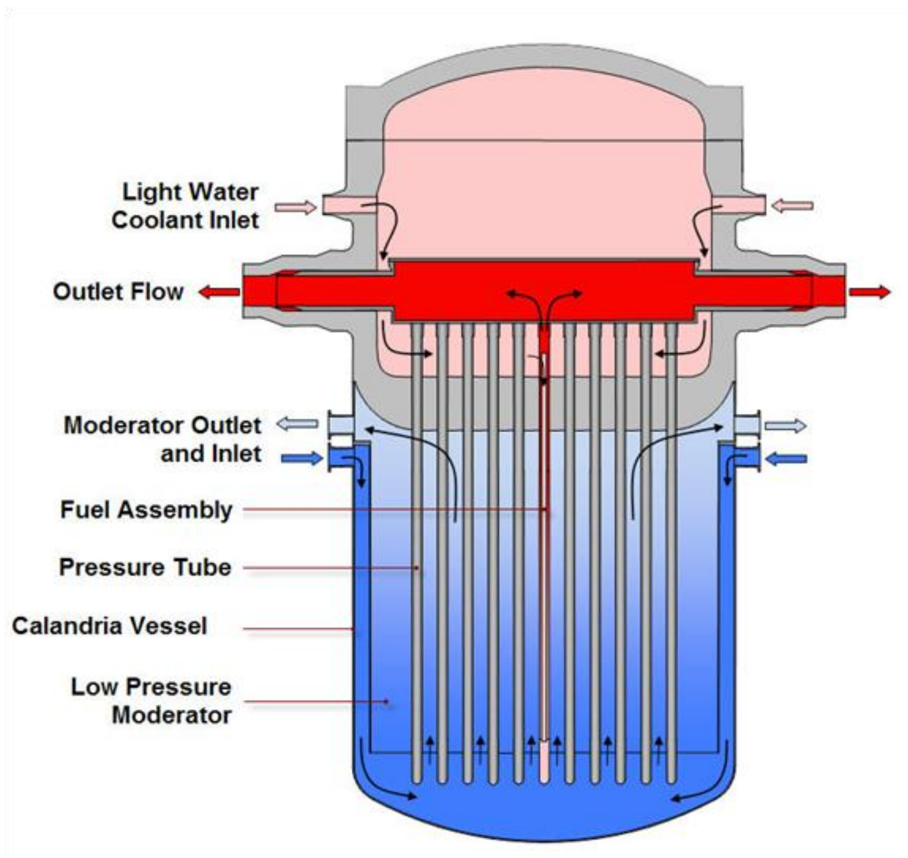

Figure 3: Schematic of the Canadian SCWR core [1].

2.1 Calandria

The calandria is a separate vessel, attached to the inlet plenum, which houses the pressure tubes and the reactivity control and shutdown mechanisms. The calandria is filled with a heavy water moderator, with a cover gas system (to remove radiolysis products) in the active moderator cooling system to minimize the free surface area. The outside surface of the calandria will be exposed to the ambient conditions within the containment building, while the inside surface will be exposed to heavy water at temperatures up to 103 °C (30 °C subcooled); the calandria will be exposed to a neutron flux. The selection of suitable materials for the calandria requires only incremental increases in knowledge compared to existing CANDUs, and therefore this is not considered a critical path item, and will not be discussed further in this paper.

2.2 Fuel Channels

Each fuel channel consists of a pressure tube, a pressure tube extension, an insulator and an insulator liner. The pressure tube forms the pressure boundary between the SCW coolant and the moderator. The key enabling technology of the Canadian SCWR concept is the high efficiency channel (HEC) [23, 24], which limits heat transfer to the moderator and tubesheet by using an insulator to separate the high temperature coolant from the Zr-alloy pressure tube. The insulator liner protects the insulator against mechanical damage and contains any fragments should the insulator fracture.

The inside surface of the fuel channel (liner) will be exposed to coolant at 25 MPa and temperatures from about 350 to 625 °C, while the outside surface of the fuel channel (pressure tube) will be exposed to moderator conditions. Given its proximity to the fuel and therefore exposure to similar conditions, the liner material will likely be the same alloy as the fuel cladding. The pressure tube temperature is below 200 °C due to the direct contact of the pressure tube with the cool heavy water moderator. At this low temperature the strength and corrosion rate of Zr alloys are acceptable, allowing the use of these low neutron-absorbing alloys for the pressure tube and hence retaining neutron economy. A high strength, creep resistant Zr alloy, Excel (Zr - 3.5%Sn - 0.8%Nb - 0.8%Mo - 1130 ppm O), developed by AECL in the 1970s [25, 26, 27] is the reference pressure tube alloy, used in the annealed condition to minimize irradiation creep and growth rates.

Deuterium from the moderator, and possibly hydrogen from the coolant (depending on the extent of coolant leakage between the insulator/liner and the pressure tube) will enter the Zr-alloy pressure tube as a by-product of oxidation. The low ingress rate at the pressure tube operating temperature and the high Terminal Solid Solubility (TSS) for hydrogen in Excel [28] mean that TSS will not be exceeded for the design life of a Canadian SCWR pressure tube (assumed to be 60 years at 90% capacity factor). This eliminates one of the necessary conditions for delayed hydride cracking, the only cracking mechanism observed in Zr-alloy pressure tubes in service. The microstructure of Excel differs from that of Zr alloys such as Zircaloy-2 and Zr-2.5% Nb due to differences in composition, although there are some similarities. Although not considered critical path, work is underway to characterize the microstructural evolution of Zr-Excel alloys under irradiation [29].

Yttria Stabilized Zirconia (YSZ) has been selected as the reference insulator material because it has a low neutron cross-section, low thermal conductivity and very high corrosion resistance in SCW [30]. Stabilized zirconia has been shown to be exceptionally resistant to irradiation damage from fast neutrons and energetic ions. Limited data showed that irradiation would not significantly embrittle YSZ, at least at high temperatures [31]. No amorphization was observed in stabilized zirconia under neutron [31] or ion irradiation to high dpa values [32, 33, 34].

The development of the insulator is a critical path item for the Canadian SCWR concept. The on-going evolution of the HEC concept is being driven by safety requirements that ensure sufficient decay heat removal following a loss of coolant accident to maintain fuel cladding integrity (the "no-core melt" safety case). Licht [35, 36] performed a parametric investigation of the HEC parameters that affect heat removal, including the fuel cladding and liner tube emissivities, insulator thickness, insulator conductivity and the temperature limit of the fuel cladding.

2.3 Fuel Assembly

The fuel assembly consists of the nuclear fuel and support structure, and is inserted into the fuel channel. Coolant enters the fuel assembly through one of four ports close to the top, travels down a central flow tube to the lower end of the fuel assembly, changes direction by 180 degrees, and then flows upwards through the fuel elements. The coolant temperature remains below the critical temperature as it passes through the flow tube, and is heated while flowing upwards, passing through the critical temperature and discharged directly into the outlet plenum at 625 °C.

The Canadian SCWR will employ batch refuelling, and each fuel assembly will reside in the core for 3.5 years (3 cycles of 425 d length = 3.5 y [37]). An advantage of the pressure tube SCWR concept is that all components exposed to the combination of irradiation and exposure to the SCW coolant can be removed from service when the fuel assembly has reached end-of-life. This significantly reduces the materials requirements (corrosion and SCC resistance, irradiation damage resistance), although it would increase waste disposal costs.

The fuel assembly components are subjected to the most challenging conditions in the SCWR, a combination of high temperature, high pressure and irradiation. An example of the variation of the coolant and cladding temperatures as a function of distance along the fuel channel along the fuel side of the channel is shown in Figure 4. Also shown are the variations in bulk and surface coolant densities; the surface density was calculated from the cladding temperature assuming that the properties of SCW at a surface are the same as those in the bulk. Svishchev et al. [38] have recently shown that this is not correct; the surface imposes a structure on the adjacent water that is different than that of the bulk. As it is the surface density that will affect corrosion and corrosion product deposition, the implications of this change in structure must be assessed.

It is important to note that many of the effects of irradiation on materials have strong temperature dependencies. Since the cladding temperature varies significantly along the length of the fuel channel, the relative importance of various phenomena may change. For example, some types of irradiation damage may be more of an issue near the core inlet, as rapid diffusion at the high cladding temperatures near the core outlet may lead to annealing of the damage in this region.

Figure 5 shows the neutron energy spectrum at the flow tube and at the cladding on inner and outer fuel elements in the Canadian SCWR. Using these data, calculations were performed using the SPECTER code [39] for several alloys of interest; for Alloy 800HT, dpa and He production values of 9 dpa and 50 appm He respectively were obtained.

The central flow tube presents no significant material challenges as the operating conditions are within the existing knowledge base. However, selection of the fuel cladding material is clearly a critical path item.

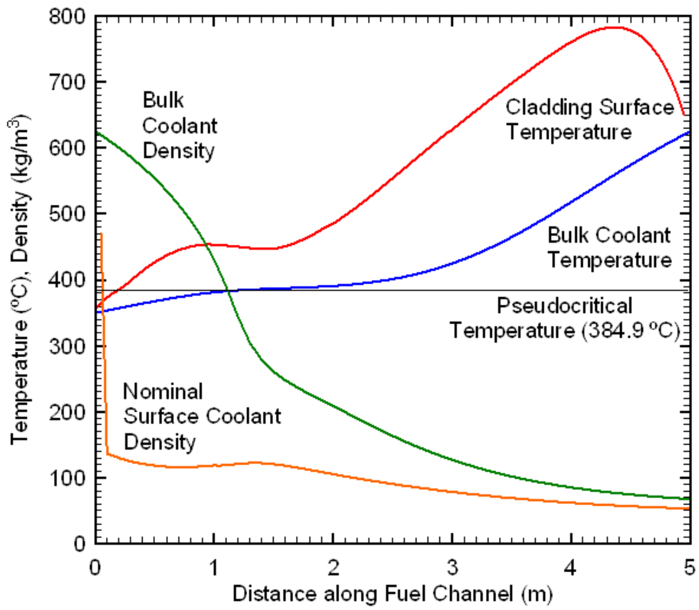

Figure 4: Variation of the coolant temperature and the cladding temperature as a function of distance along the fuel channel as it passes over the fuel.

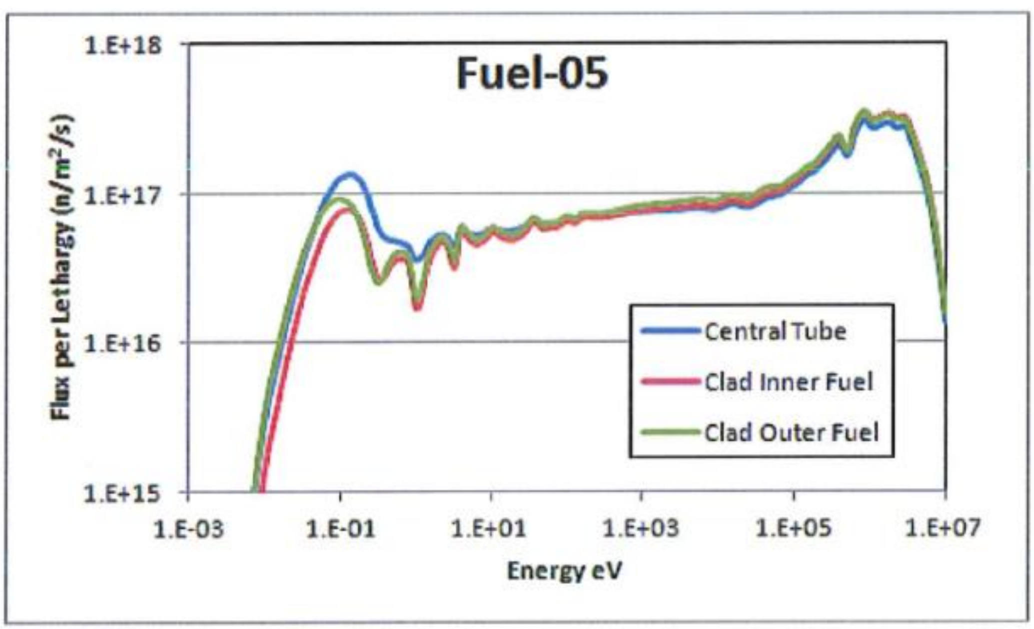

Figure 5: Neutron energy spectra at the SCWR Fuel-05 axial location for inner and outer fuel and central flow tube locations.

2.3.1 Fuel Cladding

A free-standing, internally pressurized fuel cladding has been proposed by the EU [22] and Japan [21]. In these designs, He would be added to the fresh fuel rods to give an initial He pressure of about 8.5 MPa, so that the maximum pressure inside the fuel rods would be close to 25 MPa at the end of a cycle. The analyses performed for the Japanese SCWR were adopted for the HPLWR [22]. As a result, a yield strength of 160 MPa at 550 °C and 150 MPa at 650 °C, and a 20,000 h creep strength of 120 MPa at 600 °C became cladding material requirements. The maximum allowable reduction of wall thickness by corrosion for the High Performance Light Water Reactor fuel cladding was 140 μm after 20,000 h.

In current CANDU reactor fuel designs, the Zr-alloy fuel cladding is designed to collapse into contact with the UO_2 fuel under reactor coolant conditions [40]. The as-fabricated diametrical clearance between the UO_2 pellet stack and the sheath is controlled within a range that:

1. Prevents the formation of longitudinal ridges in the sheath;
2. Facilitates pellet loading during fuel element manufacturing; and,
3. Accommodates some of the pellet diametrical expansion and minimize sheath strain.

Before pellet loading, a thin layer of graphite is applied to the inner surface of the fuel cladding to reduce the pellet-cladding interaction. The void within the fuel elements is filled (but not pressurized) with a He/air or He/inert gas mixture prior to attaching the endcaps to facilitate leak detection during fabrication and to improve pellet-to-sheath heat transfer.

In an SCWR, the relatively small gap between the fuel and cladding will shrink during normal operation due to fuel thermal expansion. The gap size for the Canadian SCWR will be chosen such that the stainless steel[2] fuel cladding will collapse onto the fuel pellets, allowing the 25 MPa pressure on the outside of the fuel cladding to be supported by the fuel pellets. As a result, the high-temperature mechanical strength and creep resistance properties of the fuel cladding become secondary factors in material selection. As the fuel proceeds through its lifetime in the core, fuel swelling due to irradiation will introduce a tensile hoop stress to the cladding, which will have to be considered during the characterization of the SCC resistance.

The idea of a collapsible stainless steel fuel cladding is not new; such a concept was evaluated in tests performed at the Vallecitos BWR in 1961 [41] using superheat fuel element SH-4B. Among the design objectives of this test were to decrease the amount of stainless steel in the cladding to improve neutron economy, and to examine the concept of supporting the metallic cladding against the external pressure by means of the pellet strength. The fuel element was constructed with a 0.71 mm thick 304 stainless steel cladding, and exposed to superheated steam at 6.9 MPa for a total of 617 h, including 80 h at a maximum steam temperature of about 480 °C. The test was terminated due to the occurrence of fuel defects, attributed to SCC. A key conclusion of the test was that a fuel element with a thin-walled cladding supported by the UO_2 fuel could be made workable for nuclear superheat applications.

Considerable operating experience also exists on the use of stainless steel fuel cladding in BWRs and PWRs [42]. At the Beloyarsk NPP, Kh18Ni10T and EI-847 stainless steels were used [12].

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[2] Or possibly a nickel-based alloy.

A key factor in cladding material selection is the wall loss due to corrosion; this must be characterized and sufficient margin added to the fuel cladding thickness such that at the end of design life the wall thickness is above the minimum required thickness to ensure mechanical integrity. Prediction of the wall loss requires the measurement of the end-of-life corrosion penetration (in μm) or weight loss (mg/dm^2). As noted above, the the maximum allowable reduction of wall thickness by corrosion for the High Performance Light Water Reactor fuel cladding has been specified to be 140 μm after 20,000 h.

In this regard, the typical practice of reporting weight change measured in corrosion tests is of no value unless it can be shown that the weight gain can be converted to a wall loss. This requires demonstration that there is no dissolution of the corrosion film into the coolant and no spalling of the oxide. In the absence of test data spanning the expected in-service life of the cladding under relevant conditions, some extrapolation is required to obtain the wall loss after 3.5 y at 850 °C. Extrapolation in time requires good data over moderate time periods (about 2000 h) to allow fitting of the data to typical corrosion kinetics (linear, parabolic, logarithmic). There are no autoclaves capable of performing tests at 850 °C at 25 MPa. However, a number of authors [43, 44, 45] have shown that corrosion rates in SCW at 25 MPa (high-density SCW) and in low pressure superheated steam (low-density SCW) are similar, allowing superheated steam to be used as a surrogate for high-density SCW. Canada's test program includes corrosion and SCC testing at temperatures up to 800 °C in low pressure steam. A second key factor is the oxide thickness, as the formation of a thick oxide layer will affect the thermalhydraulics by changing the dimensions of the gaps between adjacent fuel elements and changing the heat transfer from the cladding to the coolant.

A crossover at the upper end of the flow tube directs coolant to the flow tube and fixes the flow tube within the fuel assembly liner. As well as directing the flow of the coolant through the fuel assembly, the flow tube and outlet tube are designed to limit heat transfer between the steam and coolant. This may be accomplished by a simple double wall construction, or these components may be coated with a thermal barrier material. As the thermal stresses on this component could be significant without careful design, this component is a critical path item.

3. Conclusions

There are three components on the critical path for materials selection for the Canadian SCWR: the fuel channel insulator, the fuel cladding, and the coolant crossover. Canada's Generation IV National Program [46, 47], now in its second, 4-year, phase, has devoted a significant effort to developing the required materials test facilities. Building on earlier work under the CANDU X program, a large, ongoing program of corrosion and SCC testing, including fundamental studies aimed at developing the mechanistic understanding required to develop predictive models is in place. The program also include projects to characterize mechanical properties, improve insulator performance, characterize the behaviour of the Excel pressure tube alloy, and develop corrosion resistant and thermal barrier coatings. The Phase II program also includes corrosion and SCC testing in low pressure, high temperature 'steam' at temperatures up to 800 °C.

A significant knowledge gap exists with respect to the influence of the slowly evolving microstructure of stainless steels due to thermal ageing (diffusion and secondary phase precipitation, particularly involving grain boundaries) [48, 49, 50]. Generally, austenitic stainless steels that have no δ-ferrite remain austenitic from room temperature up to about 550 °C, above which the alloy can decompose from a solid solution into various carbide or intermetallic precipitate phases plus a more stable austenite phase [51]. As the peak cladding temperature of the Canadian SCWR concept is above 550 °C for more than half of the length of the fuel channel (Figure 4), these effects could be significant. Thermal ageing is known to create a sensitized microstructure in which chromium carbide ($M_{23}C_6$) precipitates on the grain boundaries, rendering them more susceptible to corrosion [52]. However, the influence of other intermetallic phase whose formation requires longer ageing times, such as the chi, laves and sigma phases, on corrosion resistance has received little attention. Almost all alloys tested to date were assessed using as-fabricated materials, without much consideration of a slowly evolving microstructure.

The major knowledge gap for fuel cladding selection for the Canadian SCWR is the lack of data on irradiated materials. The Japanese performed irradiations of a number of alloys (316 SS, various modified versions of 310 SS, and Alloy 690) in the Japanese Materials Testing Reactor and the JOYO Experimental Fast Reactor at temperatures up to 700 °C. Many of the alloys of interest have also been irradiated at relevant temperatures in support of the development of other reactor concepts, e.g., fast reactors. These data need to be carefully evaluated for their relevance to the Canadian SCWR conditions (temperature, fluence, neutron spectrum, etc.) and a test plan developed to perform targeted experiments to obtain any required data.

Acknowledgements

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