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Supercritical water-cooled reactor materials – Summary of research and open issues

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A B S T R A C T

The Supercritical Water Reactor (SCWR) is one of the six reactor concepts being investigated under the framework of the Generation IV International Forum (GIF). Research on materials and chemistry for supercritical water-cooled reactors dates back to the 1960s when a number of reactor concepts using water at supercritical temperatures but sub-critical pressures (nuclear steam) were studied. There is also significant experience available from the operation of supercritical fossil-fired power plants. In this paper, the materials requirements of the various SCWR concepts are introduced, with a focus on the European Union pressure vessel concept and the Canadian pressure tube concept. The current understanding of the key materials degradation issues is reviewed, and knowledge gaps identified.

1. Introduction

The idea of using a supercritical water (SCW) coolant in a water-cooled reactor dates back to the 1960s (Dollezhal et al., 1965; Wright and Paterson, 1966), although no reactor operating at both supercritical temperature and supercritical pressure was ever built. More recently, two types of SCWR concept have evolved from existing light water reactor (LWR) and pressurized heavy water reactor (PHWR) designs: (a) a number of designs (Oka and Koshizuka, 1993; Schulenberg et al., 2009; Bae et al., 2007) consisting of a large reactor pressure vessel containing the reactor core (fueled) heat source, analogous to conventional pressurized water reactor (PWR) and boiling water reactor (BWR) designs (Fig. 1); and (b) designs with distributed pressure tubes or channels containing fuel bundles, analogous to conventional CANDU® and RBMK[1] nuclear reactors (Yetisir et al., 2013) (Fig. 2). The balance-of-plant is typically a direct-cycle design (Fig. 3) and the out-of-core portions of both concepts are similar to those found in existing fossil-fired generators. There is significant industry experience with the use of SCW in non-nuclear power generation, with about 268,944 MWe (462 units) of installed capacity in coal-fired SCW power plants worldwide (Viswanathan et al., 2004) as of 2004.

1.1. Fossil and SCWO operating experience

There are a number of excellent summaries of the work that has been carried out in support of materials development for supercritical and ultrasupercritical fossil-fired power plants, including a recent paper by Wright and Dooley (Wright and Dooley, 2010).

A nuclear reactor core is significantly different from a fossil-fired boiler. The latter contains a large number of relatively thick-walled (~6–12 mm thickness) fire tubes that circulate water on the inside and are heated from the outside. In a water-cooled reactor, the need for neutron economy dictates the use of a thin fuel cladding to contain the nuclear fuel; the water is circulated over the outside of the fuel cladding. Typical fuel cladding thickness in an SCWR will be in the range 0.4–0.6 mm, providing little corrosion margin. This 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. In addition, irradiation of the coolant and in-core materials leads to various forms of degradation that are not encountered in fossil-fired plants (FFPs).

Oxide films on FFP boiler tubes, formed by corrosion of the tube material and by deposition of corrosion products transported from

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[1] ®CANDU – CANada Deuterium Uranium, is a registered trademark of Atomic Energy of Canada Limited (AECL). RBMK – Reactor Bolshoy Moshchnosty Kanalny.

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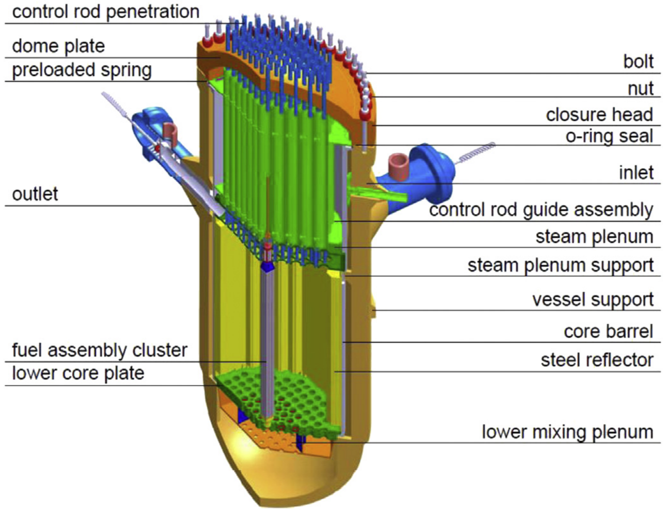

Fig. 1. Schematic of the High Performance Light Water Reactor (HPLWR) core design concept (Schulenberg, 2013).

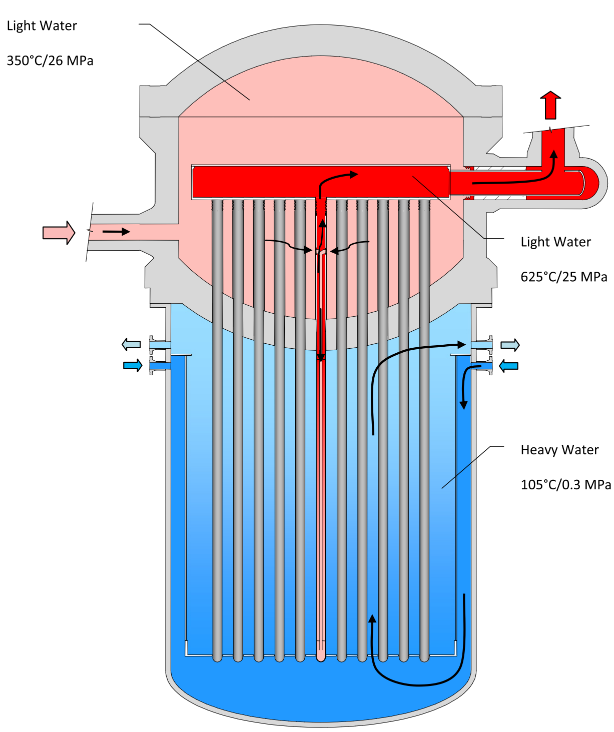

Fig. 2. Schematic of the Canadian SCWR core (Yetisir et al., 2013).

the feedtrain, can be several hundred micrometers thick, which would be unacceptable on a SCWR fuel cladding. Such thick deposits could result in a) overheating of the cladding surface or underdeposit corrosion, leading to fuel failures, b) changes in reactivity in the core (crud-induced power shifts[2]), and c) increased radiation fields on out-of-core piping.

In addition to the operating experience and research in support of fossil-fired SCW plants (FFSCWPs), a large amount of data on materials degradation in SCW was acquired during the development of Supercritical Water Oxidation (SCWO) processes. While the chemistry conditions in these tests are generally not of direct relevance to a SCWR, typically being acidic with high concentrations of aggressive species such as chloride, these data do provide some insights into the key parameters affecting corrosion phenomena in SCW (Kritzer, 2004). However, despite information available from current reactor designs, modern boiler technologies and research in support of SCWO, significant gaps still exist in our understanding of the behavior of materials under proposed SCWR operating conditions.

1.2. Water chemistry

The key water chemistry issues for SCWR concepts have been summarized (Guzonas et al., 2012; Kysela et al., March 8–11; Yurmanov et al., 2010). It is important to note that many of the chemistry control practices in FFSCWPs are aimed at minimizing corrosion of the feedtrain, rather than the boiler. Two key chemistry issues have been highlighted for the SCWR. The first is the transport of corrosion products and impurities such as chloride from the feedtrain to the core. In 1960, Marchaterre and Petrick noted that “The major gap in supercritical water technology pertaining to a reactor system is the lack of information on the magnitude of the problems of deposition of radioactivity in the external system and of the build-up of internal crud under irradiation.” Burrill (2000) predicted potentially high in-core oxide deposits using deposit data obtained from FFSCWPs. However, little or no laboratory data are available to validate these predictions.

Modeling provides the only means of assessing potential corrosion product deposition under SCWR conditions in the absence of laboratory measurements. Cook and Olive (April, 2012), Olive (September 2012) and Cook and Olive (2013 September 1–5) recently modeled iron and nickel deposition in the Canadian SCWR core for scenarios including: 1) coolant saturated in the metal species of interest at the core inlet; and 2) coolant unsaturated in the metal species of interest at the core inlet. With saturated coolant, deposition started at the core inlet, reached a maximum about 1 m into the core and continued until the core outlet. For unsaturated coolant (1 $\mu g\ kg^{-1}$ dissolved Fe), deposition started roughly 1 m into the core and continued until the core outlet.

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. A study in the BONUS[3] reactor showed that wet steam containing chlorides and oxygen caused chloride-induced SCC failure of Type 304 and Type 347 stainless steels. Chloride deposition from the drying of moist steam resulted in heavy, adherent localized deposits (Bevilacqua and Brown, 1963), which in the presence of oxygen and water were conducive to severe chloride-induced SCC of austenitic steels. While Unit 2 at the Beloyarsk Nuclear Power Plant (a pressure-tube BWR with nuclear steam reheat channels) operated for many years with a typical chloride concentration of 25 $\mu g\ kg^{-1}$ with no reported negative effects (Yurmanov et al., 2010), laboratory tests showed that the stainless steel used for the channel elements (1Kh18N10T) cracked due to SCC after 144–1100 h of temperature and pressure cycling in an environment containing chloride. It was suggested that in the laboratory tests, deposition of moisture on the outer surface and subsequent evaporation may have led to chloride concentration on the surface (Emel'yanov et al., 1972). Current BWRs operate with feedwater chloride concentrations as low as 0.25 $\mu g\ kg^{-1}$ (Stellwag

[2] Also known as Axial Offset Anomaly (AOA).

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