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Pool boiling heat transfer characteristics of zircaloy and SiC claddings in deionized water at low pressure

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ABSTRACT

Since the Fukushima accident in 2011, deployment of silicon carbide (SiC) materials has been investigated as a replacement for zirconium-based cladding in light water reactor claddings. SiC cladding exhibits many advantages including an exceptionally high melting temperature, considerably slow rate of hydrogen generation from reaction with water at high temperatures, and less corrosion compared to currently used zircaloy cladding. To verify the enhanced safety margin and potential features of accident tolerance of the SiC cladding, the pool boiling heat transfer characteristics were investigated in deionized water under atmospheric pressure and compared to zircaloy-4 cladding. The SiC monolith claddings were manufactured and donated by GAMMA CTP. The zircaloy-4 claddings were heated via Joule heating using a 12 kW DC power supply while measuring the inner wall temperature. However, since the electrical resistance of SiC cladding is exceptionally high, indirect heating was applied by centering a heating element consisting of a stainless steel tube with an outer diameter of 6.35 mm inside the SiC cladding. The measured critical heat flux (CHF) of the SiC monolith cladding was approximately 63% higher than that of the zircaloy-4 cladding. Furthermore, after occurrence of the CHF, in spite of stable film boiling formation on the SiC surface, no physical damage was observed with the SiC cladding whereas the zircaloy-4 cladding experienced rapid physical degradation at the CHF and finally fragmented. High-speed video recorded at a frame rate of 1500 fps provided observation of the distinctive features of the characteristic boiling phenomena of the zircaloy-4 and SiC claddings. The hydrodynamic instability near the CHF, the transition from the CHF to film boiling, and the stable film boiling regime were captured successfully. For the first time, this study demonstrates that SiC cladding has the sustainable structural integrity with visual observation after the CHF occurrence and can be advantageous in securing a high safety margin for nuclear power reactor applications.

1. Introduction

Zirconium alloys have been widely used as fuel cladding materials in commercial light water reactors (LWR) over the past several decades. They have various advantages as a nuclear fuel cladding due to their high mechanical strength, high melting point (∼1850 °C), high corrosion resistance, and low absorption cross-section for thermal neutrons [1]. However, despite these advantages, several limitations of zirconium-based claddings have been reported. Stress and corrosion induced cracking, embrittlement by irradiation, and hydrogen pickup may potentially damage fuel rods during operation due to extended burnup [2]. In addition, the exothermic oxidation process and hydrogen generation of a zircaloy cladding can be accelerated at high temperatures under transition conditions, which makes the management of accidents more difficult [3].

In fact, approximately 1000 kg of hydrogen could have been produced from complete reaction of all of the claddings with high temperature steam in the Three Mile Island-Unit 2 (TMI-2) accident [4]. Although this did not occur in the TMI-2 accident, if the generated hydrogen was accumulated locally in the reactor system, it could detonate the reactor system, which was the case in Fukushima Daiichi nuclear power plant accident [5]. Recent numerical analysis using the MELCOR system code has also confirmed that the zircaloy and steam reaction is a significant heat source that needs to be managed for reactor vessel cooling in severe environments [6]. As such, demands for a higher safety margin and accident tolerant cladding have recently arisen.

As a strong candidate as an accident tolerant cladding material, silicon carbide (SiC) has received a lot of attention. SiC is the primary candidate for GFR, LFR, and VHTR, which are generation

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IV (Gen-IV) nuclear reactors [1,7]. SiC cladding exhibits noticeable advantages including its high thermal conductivity, high corrosion resistance, less waste disposal restrictions, and low neutron absorption cross-section [7]. Most of all, one remarkable advantage of adopting SiC cladding is its high melting point and the considerably lower rate of hydrogen generation in severe accident environments such that the enough amount of hydrogen explosion can be avoided before an accident management begins.

However, despite the reported advantages, important issues in using SiC cladding as a new nuclear fuel cladding remain unresolved. As practical issues, the manufacturability and cost competitiveness of SiC cladding have not been industrially achieved. Most importantly, there is a lack of phenomenological understanding of SiC cladding in terms of its reactor water chemistry and thermal–hydraulic operating conditions [1].

To address these important issues, several studies of SiC material have been conducted. Snead et al. evaluated the properties of SiC material under non-irradiated and irradiated conditions [8]. In their work, comprehensive data regarding SiC are presented, which is extremely valuable for nuclear applications. Carpenter investigated the irradiation performance of various SiC claddings in his Ph.D. dissertation work [9]. In the experiments, monolithic SiC and TRIPLEX tubes were tested under LWR conditions and the effects of irradiation and structural design on the SiC tubes were carefully investigated. From the experimental results, no significant creep and elastic deformation were observed over a large temperature range. However, SiC material experienced corrosion with water under LWR conditions. As a result, the strength of the SiC cladding was observed to be degraded. Carpenter discovered that these disadvantages could be improved using careful manufacturing techniques with high purity β-SiC and SiC fibers. He concluded that, in spite of its weaknesses, the SiC composite can be a feasible candidate for nuclear fuel cladding to overcome current issues of fuel claddings materials. More recently, Lee et al. experimental assessed the SiC cladding oxidation rates under relevant conditions with loss-of-coolant accidents (LOCA) in order to understand its key failure mechanisms [3]. The experimental results showed noteworthy different oxidation behaviors of SiC and zircaloy-4. The SiC claddings underwent weight losses in steam while the zircaloy-4 experienced weight gains. They concluded that a reaction between silica (SiO_2) and water vapor (H_2O) caused the weight losses due to outward diffusion of the SiC product species. However, the oxidation rates of SiC were approximately three orders of magnitude slower than those of zircaloy-4. A more relevant study to the current investigation is the experimental work of aus der Wiesche et al., which involved the pool boiling heat transfer of diamond and SiC plate heaters [10]. The experimental results showed that thermal wall properties contributed to higher heat transfer coefficients of the diamond heater than those of the SiC heaters in the nucleate boiling regime. On the other hand, it was reported that the heating wall properties rarely affected the natural convection mode and bubble dynamics including the bubble departure diameter and frequency during nucleate boiling. Their study provides a useful reference on the pool boiling heat transfer of a SiC heater. However, understanding of the boiling crisis on a SiC surface remains insufficient because their experiments were limited to a maximum heat flux of 500 kW/m^2 due to the premature destruction of the oxidation layer at a high heat flux. Therefore, CHF data were unavailable in their study. Unlike the work of aus der Wiesche et al., a few studies reported CHF enhancement with SiC material using SiC nanofluids [11,12]. Besides the aforementioned experimental studies, a numerical study was also carried out for the potential application of SiC cladding in existing nuclear power plants. Seo et al. investigated the effects of the deployment of SiC claddings as a replacement of zircaly-4 claddings on the thermal–hydraulic behavior of OPR1000 [13], which is a type of commercial power reactor in Korea. A large break loss-of-coolant-accident (LBLOCA) scenario was selected for the analysis using MARS code, which is a system safety analysis code developed in Korea. For simplification, the SiC cladding was modeled without changing the dimensions of the current zircaloy cladding. The simulation results showed lower peak cladding temperatures (PCT) for the SiC claddings than the zircaloy claddings in the blowdown and reflood phases. The reduced PCTs may be contributed to the higher thermal diffusivity of SiC, which enhances the safety margin during transient conditions.

Consequently, the objective of this study is to investigate the thermal–hydraulic behavior of SiC in pool boiling heat transfer including observation of CHF phenomenon. SiC monolith cladding heaters were selected and compared to zircaloy-4 cladding heaters. Pool boiling heat transfer experiments were performed in a pool of deionized water (DI water) at atmospheric pressure. The CHF and nucleate boiling heat transfer coefficients of both types of claddings were compared. Also, visualization of boiling was conducted using a high-speed video system to characterize the distinctive boiling regimes.

2. CHF correlation

Prediction of the CHF has been a crucial task in thermal–hydraulic evaluations to accurately estimate the safety margin for high power nuclear reactors. However, due to the complex thermal–hydraulic pressure, temperature, and flow characteristics, no single universal CHF prediction has been developed to date. Initiated with intuitive observations of fundamental pool boiling heat transfer, CHF prediction has a long history of development. In 1951, Kutateladge, a pioneer of the CHF field, developed a CHF correlation for the first time in the pool boiling of water on a plate heater, as shown in Eq. (1). Zuber (1959) later concretized the CHF triggering mechanism by introducing the instability theory, which states that a vapor jet velocity reaching the critical value triggers the CHF condition. Then, he determined the coefficient C in Eq. (1) to be $\pi/24$, which is the wavelength of the unstable vapor jet column repeatedly formed on the infinite plate heater. Since the pioneering work of Kutateladge and Zuber, Lienhard et al. also developed a similar CHF correlation on a plate heater with a slightly different C coefficient [14].

$$q''_{\mathrm{crit}} = C h_{fg} \rho_v^{1/2} \left[\sigma g(\rho_l - \rho_v)\right]^{1/4} \tag{1}$$

The development of CHF correlations has followed many directions depending on the heater geometry, orientation, and surface characteristics, to mention just a few factors [15,16]. In the current study, CHF with a tube geometry is of special interest and therefore, it is worth considering relevant CHF studies. In developing a CHF correlation for tube type heaters, many researchers extended the Zuber correlation as a reference frame. Sun and Lienhard (1970) developed the CHF correlation shown in Eq. (2) using the reference Zuber's prediction equation. The dimensionless radius shown in Eq. (3) was introduced to take into account the cylinder radius effect.

$$\frac{q''_{\mathrm{crit}}}{q''_{\mathrm{crit,Zuber}}} = 0.89 + 2.27\exp\left(-3.44\sqrt{R'}\right) \quad (R' > 0.15) \tag{2}$$

The dimensionless radius R' is defined as follows:

$$R' = R\sqrt{\frac{g(\rho_l - \rho_v)}{\sigma}} \tag{3}$$

Tong and Tang (1997) stated that vapor removal configuration in the vicinity of the CHF depends on the heater diameter [17]. For small cylinders of $0.2 < R' < 2.4$, Sun and Lienhard's modified correlation is described by the empirical formula shown in Eq. (4).