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Potential impact of accident tolerant fuel cladding critical heat flux characteristics on the high temperature phase of reactivity initiated accidents

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ABSTRACT

The present paper details scoping RELAP5-3D simulations to assess the potential impact of boiling heat transfer coefficients and critical heat flux (CHF) on peak cladding temperature (PCT) of key accident tolerant fuel (ATF) cladding types in a design basis accident (DBA). The accident event studied is a Hot Zero Power (HZP) Reactivity-Initiated Accident (RIA) in a Pressurized Water Reactor (PWR), so the present study is focused on bubble crowding CHF. Results are presented for the sensitivity of PCT and the duration time of film boiling to the key parameters in the boiling curve: nucleate, transition, and film boiling heat transfer coefficient, and CHF. This included a generic study of perturbations in boiling heat transfer characteristics for Zircaloy-4 cladding as well as a study which focused on the key candidate ATF cladding types. For Zircaloy-4 cladding, the simulation results show that both PCT and duration of film boiling are particularly sensitive to the value of CHF. We also compared the performance of FeCrAl claddings and SiC/SiC claddings with Zircaloy-4 cladding under the same RIA scenario. A 10% CHF enhancement for FeCrAl cladding can significantly reduce the PCT of FeCrAl claddings. Lower irradiated thermal conductivity of SiC/SiC claddings does cause a high temperature gradient in the radial direction.

1. Introduction

After the Fukushima accident, enhancing the accident tolerance of fuel and cladding materials in light water reactors (LWRs) became a research focus in the United States (Goldner, 2012) and subsequently across the globe. The U.S. Congress directed the Department of Energy Office of Nuclear Energy (DOE–NE) to develop accident tolerant fuel (ATF) as a part of the Fuel Cycle Research and Development (FCRD) Advanced Fuels Campaign (AFC) (Barrett et al., 2015). The overall goal is to identify alternative fuel and cladding technologies which enhance the safety, sustainability, and economic competitiveness of the U.S. commercial power reactor fleet (Barrett et al., 2015; Rebak et al., 2016). The goal of the AFC is insertion of a lead fuel rod or lead fuel assembly in a commercial reactor by 2022 (Barrett et al., 2015; Rebak et al., 2016).

Several material candidates are being considered as potentially accident tolerant alternatives to zirconium alloy cladding, including oxidation-resistant Fe- and Cr-based alloys, silicon carbide fiber-reinforced SiC ceramic matrix composites (SiC/SiC), and refractory alloys (Zinkle et al., 2014). Specifically, ferritic iron-chromium-aluminum (FeCrAl) alloys and nuclear grade SiC/SiC composites are two primary cladding candidates that are being considered to replace the zirconium alloy fuel cladding for current LWRs (Brown et al., 2017; Rebak et al., 2016). Significant research and development efforts are underway to characterize both of these candidates for cladding applications.

FeCrAl alloys exhibit excellent high temperature oxidation resistance ($\sim$1000x slower) (Pint et al., 2015) and comparable or better corrosion resistance (Terrani et al., 2016) comparing to Zr-based cladding, which demonstrate a near-term and relatively simple solution for providing enhanced safety to LWRs with minimal thermal-hydraulic design changes (Rebak et al., 2016; Terrani et al., 2014b). The formation of a protective Al_2O_3 layer in high temperature steam promotes a slow oxidation rate under high

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Nomenclature

g	gravitational acceleration (m/s^2)
H_{lg}	heat of vaporization, J/kg
h	heat transfer coefficient, W/(m^2·K)
q	heat flux, W/m^2
li-cnv	single-phase liquid convective heat transfer
sub-nb	subcooled nucleate boiling heat transfer
sat-nb	saturated nucleate boiling heat transfer
sub-tr	subcooled transition boiling heat transfer
sat-tr	saturated transition boiling heat transfer
sub-fm	subcooled film boiling heat transfer
sat-fm	saturated film boiling heat transfer
vp-cnv	single-phase vapor/gas convective heat transfer

Greek symbols

β	dynamic receding contact angle, degrees
θ	contact angle, degrees
ρ	density, kg/m^3
σ	surface tension, N/m
ϕ	heater surface angle with horizontal, degrees

Subscripts

CHF	critical heat flux condition
fb	film boiling
g	vapor
l	liquid
nb	nucleate boiling
tr	transition boiling

temperature steam environments and thus provides multiple benefits during severe accident (Pint et al., 2013; Zinkle et al., 2014). Moreover, utilization of FeCrAl alloys can also enhance the burst margins during beyond design basis accident due to the superior high temperature strength and sufficient ductility compared to zirconium alloys (Rebak et al., 2016; Yamamoto et al., 2015). However, there are several issues that need to be resolved before full implementation of FeCrAl as a cladding material (Rebak et al., 2016), including the tritium release issue (zirconium acts as a getter material for tritium) and fuel cycle economics impact caused by parasitic neutron absorption (George et al., 2015; Terrani et al., 2014b). Increasing the fuel enrichment or reducing the cladding thickness could effectively compensate for the neutronic penalty, although with a modest impact on natural resource utilization (Seo et al., 2016).

SiC – based materials, in particular SiC/SiC composites, have also been investigated as potential ATF cladding materials (and also as channel box for boiling water reactors (Yueh and Terrani, 2014)) due to their low neutron absorption (Matsumiya et al., 2015), high strength and irradiation stability (Katoh et al., 2014) and high temperature oxidation resistance in air and steam (George et al., 2015; Katoh and Terrani, 2015; Terrani et al., 2014a; Zinkle et al., 2014). Hoop and axial strengths of the SiC – based cladding can be controlled and balanced by adjusting the tube fiber structure to create pseudo-ductile mechanical behavior (Deck et al., 2015). However, attention is needed to assess design and reactor performance and safety characteristics of SiC/SiC related with manufacturing, normal operation, and off-normal events, such as the hermeticity issue caused by micro-cracking (Deck et al., 2015), lower irradiated thermal conductivity (Snead et al., 2007), and the hydrothermal corrosion in LWR coolant environments (Stempien et al., 2013; Terrani et al., 2015). The high temperature gradients through the cladding is expected to cause significant stress due to the thermal expansion and irradiation-induced, temperature–dependent swelling (Ben-Belgacem et al., 2014). The implementation of SiC/SiC–based cladding in LWRs will require design of optimized composite structures, scalable and cost-effective fabrication methods, and sufficient characterization of the material being produced (Deck et al., 2015).

The reactor performance and safety characteristics of the new fuel system with ATF claddings must be assessed relative to that of the UO_2–Zircaloy system under normal, operational transient, off-normal, design-basis accident, and beyond design-basis conditions (Brown et al., 2015). Understanding the impact of a RIA is very important for candidate fuel and cladding concepts with new materials (Brown et al., 2017). With respect to the magnitude of the reactivity insertion, the most significant RIA would occur at HZP conditions (OECD Nuclear Energy Agency, 2010). The HZP condition is a stage when operational pressure and temperature is reached but the power remains low (Fritz, 2013). With respect to potential fuel and cladding damage, however, conditions other than HZP could be more challenging (OECD Nuclear Energy Agency, 2010). For RIA events, the thermal hydraulic conditions may undergo a rapid transition from forced convection cooling to critical heat flux conditions. This rapid rise in heat flux ultimately leads to film boiling conditions at the surface of the fuel cladding resulting in high cladding temperatures and eventually the collapse of the surrounding vapor blanket, with consequent cladding oxidation (MacDonald et al., 1980). For zirconium–based cladding, waterside corrosion of the cladding yields ZrO_2 and coolant/-cladding chemical reaction during film boiling produces surface layers of oxygen–stabilized α-Zr and ZrO_2 (Ohnishi et al., 1984). Though several experimental programs have been performed to measure fuel performance phenomena under RIA conditions, the time–dependent thermal hydraulic conditions at the cladding surface remain an area of high uncertainty (OECD Nuclear Energy Agency, 2012) and thus a primary target for data collection intended for fuel performance model validation.

In many countries, the cladding failure limit for RIA is based on the Nuclear Regulatory Commission (NRC) Standard Review Plan (NUREG-800), which includes separate PWR and Boiling Water Reactor (BWR) criteria for both high cladding temperature failure and pellet-to-cladding mechanical interaction (PCMI) failure mechanisms (Clifford, 2007; OECD Nuclear Energy Agency, 2012). As defined in NUREG-800, "the high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure" (NUREG-800, 2007). International tests with Zircaloy cladding between 40 and 60 GWd/t indicated that PCMI failures may occur at lower energy deposition due to the reduction or elimination of the pellet-cladding gap and concomitant cladding embrittlement due to irradiation and hydrogen pickup (OECD Nuclear Energy Agency, 2012). Thus, the updated PCMI failure criteria is a function of oxidation and hydriding of the cladding (NUREG-800, 2007). As a summary, during a RIA transient, fuel cladding failure may occur rapidly during the prompt fuel enthalpy rise (due to PCMI) or may occur as total fuel enthalpy, heat flux, and cladding temperature increase (NUREG-800, 2007). Brown et al. have investigated the potential effects of ATF cladding on failure limit due to PCMI at the early stage of RIA (Brown et al., 2017). The complementary