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Preliminary thermal hydraulic analysis of various accident tolerant fuels and claddings for control rod ejection accidents in LWRs

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ABSTRACT

The present study assesses the effect of various accident tolerant fuel (ATF) materials on thermal hydraulic behaviors of Light Water Reactors (LWRs) in a Reactivity-Initiated-Accident (RIA). The study focuses on the impact of ATF thermal physical properties and cladding boiling heat transfer enhancement. The results show that without the heat transfer enhancement, peak cladding temperature (PCT) can be reduced by the utilization of FeCrAl, SA3-PyC150-A and HNLS/ML-A claddings but peak fuel centerline temperature (PFCT) rises at the same time. However, the reverse situation happens to the utilization of UO_2 + BeO and UO_2 + SiC composite fuels. In addition, to identify the potential impact of nucleate and film boiling heat transfer coefficient and critical heat flux (CHF), a sensitivity analysis to these parameters is conducted. The results show PCT is particularly affected by CHF. PFCT can be reduced and heat transfer deterioration is eliminated by the application of $90UO_2$ + 10SiC assembled with all the three ATF claddings as well as $90UO_2$ + 10BeO assembled with HNLS/ML-A cladding when 10% boiling heat transfer enhancement is added.

1. Introduction

Serious concerning about enhancing fuel accident tolerant performance in light water reactors (LWRs) was aroused after the Fukushima accident which released radionuclides to environment (Saji, 2017). The immediate cause of this accident is the hydrogen explosion initiated by Zircaloy and Water Reactions in the reactor core (Kim et al., 2016). For reducing public concerns about nuclear safety and developing nuclear energy safely and efficiently, it is important to increase the reliability of fuel and eliminate the possibility of Zircaloy and water reactions. As a result, accident tolerant fuel (ATF) as a solution of this issue attracts widespread attention. In 2012, the Consolidated Appropriations Act directed that the missions of the Nuclear Energy branch of the US Department of Energy (DOE) included "give priority to developing enhanced fuels and cladding for LWRs to increase safety margin" as well as "give special technical emphasis and funding priority to activities aimed at the development and near-term qualification of meltdown-resistant, accident-tolerant nuclear fuels" (Ott et al., 2015). The primary goal of the directive is to identify alternative fuel and cladding systems for existing LWRs, which helps to improve the safety, sustainability and economic competitiveness of the present and future generations of LWRs (Ott et al., 2015; Spencer et al., 2016).

In comparison with the standard UO_2/Zr fuel rod system, two potential approaches should be achieved for ATF: fuel and cladding properties should be improved to enhance safety for design-basis (DB) as well as beyond-design-basis (BDB) events; and moreover, reaction kinetics with steam should be improved to produce less hydrogen (Chun et al., 2015; Kang et al., 2017). Several ATF fuel and cladding candidates meeting the two requirements are considered as alternates of UO_2 and Zr. Most of UO_2 alternates are composites, such as UO_2 + SiC, UO_2 + BeO, UO_2 + Molybdenum and U3Si2 + UN, while Zirconium alloy cladding alternates include FeCrAl-based alloys, SiC-based monomers or composites and refractory alloys (Zinkle et al., 2014). Significant investigations had been done to characterize the properties and performance of these ATF materials. In this paper, UO_2 + SiC fuel, UO_2 + BeO fuel, FeCrAl-based and SiC-based claddings were applied to reactor core thermal hydraulic simulation.

Silicon Carbide with high thermal conductivity, low neutron cross section, high melting point and great chemical stability is one of the most promising materials that demonstrate compatibility with UO_2 (Liu et al., 2016; Slack, 1973). In 2013, the thermal conductivity of UO_2-SiC composite fuels with different pellet size and SiC volumetric fraction was measured and compared in Yeo's study (Yeo, 2013a,b). Based on the experimented results in Yeo's paper, a further theoretical research was conducted by Liu et al. (2016) to quantify the impact of UO_2-SiC composites for a range of irradiation conditions. The improved properties and performance of UO_2-SiC composite fuels were found compared to traditional UO_2 fuel, such as decreasing fuel centerline

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temperature, increasing the gap conductivity, reducing the fission gas release, decreasing the gap/plenum pressure, facilitating a reduction in pellet cladding interaction, etc. Specially, with the increasing SiC volumetric fraction in UO_2-SiC composites, the superior performance was found. But meanwhile, it was necessary to increase UO_2 enrichment to compensate for the neutronic penalty as a result of decreasing UO_2 fraction in the fuel. In this paper, three kinds of SiC volumetric fractions were applied to reactor core simulation calculation, which were 10%, 20% and 30%, respectively.

There are a lot of the same advantages for BeO and SiC, including high melting point, low neutron absorption and good chemical compatibility with current UO_2 fuel, which also makes BeO a suitable material to combine with UO_2 to enhance fuel rod heat conductance (Liu et al., 2015). However, thermal physical properties of BeO including thermal conductivity, specific heat capacity and density, which significantly affect heat conductance are different from that of SiC (Lee et al., 2017; Zhou et al., 2015). For this reason, a different safety performance possibly exists in LWRs between UO_2-BeO and UO_2-SiC composite fuels. In general, UO_2-BeO composite fuels were expected to perform better than traditional UO_2 fuel during normal operations or accident conditions (Liu et al., 2016; Liu et al., 2015). In the same way, three kinds of BeO volumetric fractions (10%, 20% and 30%) were applied to reactor core simulation calculation in this paper.

FeCrAl-based claddings with a slow-growing alumina surface oxide formation during steam oxidation indicates it is a promising choice as ATF claddings (Cheng et al., 2012). It is protective oxide Al_2O_3 generated on the FeCrAl cladding outer surface that effectively reduces the oxidation rate under high temperature steam condition. Relative experiments involving several ATF claddings oxidation resistance to steaming or steam-H_2 environment at over 1200 °C for short time were conducted in Oak Ridge National Laboratory (ORNL), and found among these ATF cladding candidates, FeCrAl alloys with ~20Cr-5Al and Fe-Cr alloys with High Cr (≥25%) ferritic steels exhibited the superior tolerance performance in severe accident (Pint et al., 2013). Moreover, another advantage of FeCrAl-based claddings is the promotion of safety margins under BDB conditions because of the superior high temperature properties compared to zirconium cladding, such as strength, creep resistance and ductility (Rebak et al., 2016; Yamamoto et al., 2015). However, the thermal neutron absorption cross-section of iron is about 12–16 times higher than that of zirconium, which makes a substantial reactivity drop in end-of-cycle for FeCrAl cladding and subsequently reduces the cycle length by about 200 days (Terrani et al., 2014). To compensate for the neutron penalty, increasing U-235 enrichment by about 1% or reducing the cladding thickness is desirable (Seo et al., 2016).

As mentioned above, SiC can be sintered with UO_2 to compose UO_2 + SiC composite fuel. But the characteristics of SiC, such as high thermal conductivity, low-thermal neutron absorption cross-section, the low creep rate, and excellent irradiation and oxidation resistance in high temperature steam environment, also make it suitable as ATF cladding candidate (Zinkle et al., 2014). However, brittleness is the challenge to the utilization of SiC-based cladding, especially in the high temperature gradient conditions (Ben-Belgacem et al., 2014). For reducing the brittleness of SiC, chemical-vapor-deposited (CVD) monolithic SiC and SiC matrix reinforced with SiC fiber (SiC/SiC) composites may be used as the SiC-based cladding, which helps to enhance the circumferential and axial strengths and reduce the stress caused by the thermal expansion and irradiation (Deck et al., 2015).

Cladding boiling heat transfer characteristics have significant impact on reactor safety performance. If boiling happens to reactor core, cladding surface temperature is mainly determined by the boiling heat transfer regimes (nucleate boiling or film boiling). Compared to Zirconium cladding, obvious difference about boiling heat transfer characteristics including heat transfer coefficients (HTC) and critical heat flux (CHF) were found for FeCrAl-based and SiC-based claddings, which could be mainly attributed to the different wettability and roughness of the two claddings (Liu et al., 2017). The contact angle of Zirconium, FeCrAl and SiC-based claddings were measured by Ali et al., and the smaller angle was detected for FeCrAl and SiC-based claddings than zirconium cladding (Ali et al., 2017). As a result, in comparison with the Zirconium cladding, significant CHF promotion was achieved for FeCrAl-based (42%) and SiC-based (63%) claddings (Seo et al., 2015, 2016). Besides, according to the saturation pool boiling experiments (El-Genk and Suszko, 2014), the nucleate boiling HTC was also strongly influenced by surface roughness, and more than 50% enhancement was measured when the surface roughness increased by an order of magnitude. In terms of film boiling, the minimum film boiling temperature and the boiling curve was also significantly affected by surface roughness and oxidation (Bui and Dhir, 1985; Sinha et al., 2003). In general, the boiling heat transfer characteristics including CHF, nucleate boiling and film boiling HTC for FeCrAl-based and SiC-based claddings could be improved (Brown et al., 2016; Liu et al., 2017). However, the exact boiling curve of the new ATF claddings is still short in the previous literatures, and further studies are needed for it.

Above ATF materials performance and safety characteristics shall be assessed under various conditions including steady-state, operational transience, off-normal, DB, BDB, Reactivity-Initiated-Accidents (RIAs), etc. (Brown et al., 2015). Prompt criticality that may cause reactor control failure may occur when RIAs happen in the period of reactor start-up. In the full power operation condition, severe overheating may occur in the reactor core with RIAs happening, which may damage the primary loop pressure boundary. In the RIAs process, if the peak power is too high and the heat flux at the cladding surface gets to CHF point, film boiling will occur and the temperature at this site will suddenly rise, which may melt the reactor core. In this situation, ATF fuel/cladding systems are expected to perform better than traditional UO_2/Zr system.

The present study is based on the subchannel analysis codes COBRA-EN (Basile et al., 1999). The following section presents physical and calculation models, boundary and initial conditions, and ATF materials properties used in this study. In Section 3, during the accident transient a comprehensive comparison about thermal hydraulic behaviors between ATF fuel/cladding systems and UO_2/Zr system is provided. Moreover, a sensitive analysis of boiling HTC and CHF for new ATF systems is conducted to identify the potential impact initiated by the superior boiling HTC and CHF. A brief conclusion was presented in the last section. The results of this paper give an assessment for thermal and hydraulic characteristics of several ATF systems contributing to the future research and application of ATF.

2. Simulation methods and ATF materials

COBRA-EN computer code that allows "Core Analysis" and "Subchannel Analysis" for LWRs is used for thermal hydraulic analysis in this study. "Core Analysis" function that allows the analysis of an assembly of open coolant channels to give the more accurate thermal hydraulics calculation is activated here (Arshi et al., 2010), because one eighth of the reactor core is analyzed in this study instead of individual rod bundle. With the application of "Core Analysis", thermal hydraulic parameters of all channels in any time and place can be figured out, including fuel temperature, enthalpy, void fraction, coolant temperature and density, pressure drop, departure from nucleate boiling rate (DNBR), etc. (Todreas and Kazimi, 2011). These functions and advantages makes the sub-channel analysis computer codes irreplaceable in the field of reactor core analysis.

2.1. Reactor description and modeling

The specifications of reactor including geometric parameters and thermal parameters are presented in Table 1. Particularly, the reactor is composed of 157 fuel assemblies, but it is enough to calculate 1/8 of