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Estimation of coping time in pressurized water reactors for near term accident tolerant fuel claddings

Anil Gurgen, Koroush Shirvan*

Department of Nuclear Science and Engineering, Massachusetts Institute of Technology, 77 Massachusetts Avenue, Room 24-215a, Cambridge, MA 02139, USA

ABSTRACT

The Fukushima Nuclear Power Plant (NPP) accident in Japan has motivated improving the safety of current light water reactors (LWRs). Accident tolerant fuels (ATF) are being developed to enhance the safety of LWRs by tolerating loss of active cooling in the core for a longer duration compared to standard UO_2 and Zirconium-based cladding. In this work, the performances of the FeCrAl and Cr-coated ATF claddings under beyond design basis accidents (BDBAs) are modeled with thermal-hydraulics design basis code TRACE. Two models are used for high-temperature oxidation of FeCrAl: a model based on the experimental results of this work, and a model based on experimental results of ORNL's work. A 3-loop Pressurized Water Reactor model is created and the following BDBAs are simulated for this study: large break loss of coolant accident without safety injection systems, short-term and long term station blackout (SBO) without any mitigation actions. In this work, we define coping time as the initiation time that assumed mitigation actions keep the clad temperature below its melting point. The results showed that ATF claddings increase the coping time and produce significantly less hydrogen compared to Zircaloy cladding under the considered BDBAs scenarios. However, the gains in coping times were found to be marginal.

1. Introduction

Following the severe accident at Fukushima Nuclear Power Plant (NPP) in Japan, safety-related concerns about the zirconium-based claddings used in current light water reactors (LWRs) have increased. Nuclear fuel cladding plays an essential role in NPP operations by being the first barrier to protect the environment from radioactivity release. High corrosion resistance and low neutron absorption cross section make Zr-based claddings a suitable material for normal LWR operating conditions. However, when the safety functions of the plant are lost and the core is uncovered during severe accident conditions, Zr-based claddings undergo exothermic reaction when exposed to high-temperature steam. As a result, the temperature of the cladding increases, leading to accelerated core melting in the severe accident progression. The generation and accumulation of hydrogen gas with its explosive potential also challenge the ability of containment structures to contain radioactivity.

To cope with the high-temperature oxidation of Zr-based claddings, accident tolerant fuel concepts were developed. The accident tolerant fuel (ATF) concept is the fuel and cladding system that gives a longer time margin, compared to Zr-based claddings, to cope with the severe accident when core cooling systems are lost in a NPP. According to Department of Energy (Goldner Frank, 2012), the fuels with enhanced accident tolerance are those that, in comparison with the standard UO_2 – Zr system, can tolerate loss of active cooling in the core for a considerably longer time period while maintaining or improving the fuel performance during normal operations. The ATF cladding should have slower oxidation kinetics and less hydrogen generation than Zr–based claddings to alleviate consequences of cladding interaction with steam at high temperatures.

Under the ATF cladding concepts, developing new metallic alloy, ceramic composite material or applying coatings on current Zircaloy cladding have been considered. Since the current regulatory framework is familiar with metallic alloys, only the metallic-based cladding concepts pursued by the nuclear vendors in the US are considered "near term" in this study. Iron-chromium-aluminum (FeCrAl) alloy and Cr are two candidates for ATF cladding with better oxidation resistance compared to Zr-based claddings (Badini and Laurella, 2001). FeCrAl alloys (commercially known as Kanthal) are widely used for many high-temperature applications. The US Global Nuclear Fuel (GNF), has recently inserted FeCrAl cladding in a commercial nuclear power plant as part of the DOE ATF program (N. E. International, 2018). Beyond current LWRs, FeCrAl cladding has also been proposed to be used for many reactor technologies including integral inherently safe LWR

* Corresponding author.
E-mail address: kshirvan@mit.edu (K. Shirvan).

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(Bojan Petrovic, 2014) and lead-bismuth cooled reactors (Lim, 2013). Chromium plating has also been widely used to improve the corrosion resistance of structures. Currently, both AREVA (Bischoff, 2016) and Westinghouse (Maier, 2018) are pursuing Cr-coated cladding to be inserted in commercial reactors in the US as part of the DOE ATF program. The oxide layers of Cr_2O_3 and Al_2O_3 have better capability to prevent diffusion of steam, hydrogen and oxygen compared to zirconium-oxide (ZrO_2) layer. Thus, in the event of a severe accident, FeCrAl and Cr would tend to generate heat and hydrogen gas at a much slower rate compared to Zr-based claddings.

It is important to have a reliable methodology to assess performance enhancements of ATF claddings compared to zirconium claddings under different accident conditions. Relevant to FeCrAl, its system level design basis accident (DBA) performance has been studied by Brookhaven National Laboratory (Cheng, 2014) and Oakridge National Laboratory (ORNL) (Ott et al., 2014) for a PWR and a BWR design respectively using the TRACE code. For beyond DBAs, Idaho National Laboratory (Merrill et al., 2017), ORNL (Ott et al., 2014) and University of Wisconsin (Jun et al., 2017) have modeled the FeCrAl performance using MELCOR. The FeCrAl oxidation kinetics in all cases were derived from experimental work by ORNL (Pint et al., 2015a). Since MELCOR was designed to model the severe accident progression after melting of fuel, currently there are no work in literature that use best estimate approach of TRACE-type codes to simulate the BDBA progression up to point of melt for both FeCrAl and Cr-Coated cladding.

In this work, a Pressurized Water Reactor (PWR) was modeled using the system level DBA code, TRACE. Thermophysical and oxidation properties of FeCrAl and Cr-coating were implemented as ATF claddings in TRACE. High-temperature steam oxidation tests were performed on FeCrAl cladding specimens using high-temperature steam oxidation facility at MIT. In these tests the specimens were heated rapidly, which does not allow sufficient time for the protective alumina layer to form on the cladding surface. Thus, this will provide a lower (conservative) bound for FeCrAl high temperature steam oxidation performance. Then the ATF claddings were subsequently simulated under BDBAs and their performance was compared to Zircaloy cladding. The selected BDBAs for this work cover wide range of time-scales for an accident progression: large break loss of coolant accident (LB-LOCA) without safety injection systems, short-term station blackout (SBO) without any mitigation actions from the beginning and long-term SBO with secondary flow for the first 24 h and the no mitigation actions afterwards. As discussed in Section 2, in this work, we define coping time as the initiation time that assumed mitigation actions keep the clad temperature below its melting point. This was chosen as a reasonable limit based on current regulatory constraints and the limitation of the TRACE code in modeling fuel meltdown.

2. Methodology

2.1. High-temperature steam oxidation facility

High-temperature steam oxidation (HTSO) facility was built in 2015 at MIT to perform high-temperature steam oxidation experiments on cladding materials (Guenon, 2016; Daines, 2016). The facility consists of a boiling pool, a quartz tube inside which steam passes and three stage heaters that heat the steam from outside of the quartz tube (see Fig. 1). Steam flow rate is driven by the 1800 W immersed heater whose input power is controlled through a DC power supply, thereby allowing for accurate control of the steam flow rate. The desired steam flow rate is calibrated before each test. Testing is performed by suspending the sample in the last furnace as the steam passes across. The last furnace temperature is controlled by OMEGA B-type thermocouple on the outside of the quartz tube with accuracy 3.0 °C at 1200 °C. Since there is no thermocouple inside the quartz tube in the last furnace, the oxidation temperature is calibrated by oxidizing multiple Zircaloy-4 (Zirc-4) samples since Zirc-4 follows Cathcart-Pawel correlation (Cathcart et al., 1977) used by Nuclear Regulatory Commission (NRC) at high temperatures. Uncertainty in parabolic rate constant of Cathcart-Pawel correlation is reported to be 2.5% by the authors (Cathcart et al., 1977). These uncertainties along with the uncertainty in weight gain measurements of Zirc-4 sample propagate to a total uncertainty of 4.2 °C. HTSO facility can have oxidation temperature up to 1500 °C and maximum steam flow rate of 15 g/min.

All the samples are ultrasonically cleaned with solvents, their dimensions are measured with a digital caliper and their weights are measured with digital chemical balance before placing to HTSO facility (Avincola, 2013). The design of the HTSO facility exposes the samples to very rapid temperature ramp rate. Basic heat balance estimates that the samples reach oxidation temperature in less than one second. The post-test SEM/EDS analysis of the samples is performed with Jeol JSM-6610LV scanning electron microscope.

2.2. TRACE system code

TRACE is a best-estimate thermal-hydraulics (TH) code (U.S. Nuclear Regulatory Commission, 2012) and it is the NRC's flagship TH modeling and safety analysis tool. TRACE has been designed to perform best-estimate analyses of LOCAs, operational transients and other accident scenarios in LWRs. It can also model phenomena occurring in experimental facilities designed to simulate transients in reactor systems. TRACE takes a component-based approach (U.S. Nuclear Regulatory Commission, 2012) to modeling a reactor system. Each physical piece of equipment in a flow loop can be represented as some type of component, and each component can be further nodalized into some number of physical volumes (also called cells) over which the fluid, conduction, and kinetics equations are averaged. The number of reactor components in the problem and the manner in which they are coupled is arbitrary. Reactor hydraulic components in TRACE include PIPEs, PLENUMs, PRIZERs (pressurizers), PUMPs, HEATRs (feedwater heaters) and VALVEs. HTSTR (heat structure) components model fuel elements or heated walls in the reactor system and are available to compute two-dimensional conduction and surface-convection heat transfer in Cartesian or cylindrical geometries. POWER components are available as a mean for delivering energy to the fluid via the HTSTR or hydraulic component walls. FILL and BREAK components are used to apply the desired coolant-flow and pressure boundary conditions, respectively, in the reactor system to perform steady-state and transient calculations. The code's computer execution time is highly problem-dependent and is a function of the total number of mesh cells, the maximum allowable time step size, and the rate of change of the neutronic and TH phenomena being evaluated. The stability-enhancing two-step (SETS) numerics in hydraulic components allows the material Courant limit to be exceeded. This allows very large time steps to be used in slow transients (U.S. Nuclear Regulatory Commission, 2012).

3. Material properties

In this section, the ATF cladding material properties that are required for TRACE model are discussed. This work utilizes FeCrAl and Cr-coating as candidate ATF materials, whose thermophysical properties and oxidation models are not directly available in TRACE. As a result, another contribution of this work is to include these necessary properties and models in the TRACE source code.

3.1. Zirc-4 properties

TRACE has Zircaloy cladding as the default cladding material (U.S. Nuclear Regulatory Commission, 2012) and it assumes the properties of zirconium-based alloys; Zircaloy-2, Zircaloy-4, Zirlo, M5, E100, etc. to be identical. Thermophysical properties of Zircaloy are based on the correlations of MATPRO (Hagerman et al., 1980) and FRAPTRAN (Siefken et al., 2001). For metal-water reaction properties, TRACE has

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