PCI analysis of Zircaloy coated clad under LWR steady state and reactor startup operations using BISON fuel performance code

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ABSTRACT

Research following the earthquake and subsequent tsunami that devastated the Fukushima Daiichi nuclear power plant, causing severe damage to the reactor cores, has led to the development of accident tolerant fuel designs. Accident tolerant fuels are described as a fuel form which exhibits improved material response, when compared to traditional uranium dioxide fuel clad by a zirconium alloy, during accident (e.g., loss of coolant accident or reactivity insertion accident) while maintaining or exceeding normal reactor operational expectations. One of the primary goals of accident tolerant fuel concepts is to reduce cladding corrosion by either replacing traditional Zircaloy cladding with a new material or by applying a coating to the outer surface of the cladding with enhanced resistance to corrosion at high temperatures. However, while coating the clad may provide an increase in high temperature corrosion resistance, it also changes the mechanical state of the cladding and potentially creates alternate failure mechanisms during reactor operations or accident conditions. This research has primarily focused on the pellet-cladding interaction/mechanical response of FeCrAl (Iron-Chromium-Aluminum alloy) coated cladding during light water reactor steady state and reactor startup operations. Using the MOOSE-based, finite-element fuel performance code BISON and commercial pressurized water reactor data obtained from Diablo Canyon Unit 2, a preliminary analysis was performed to evaluate the performance of FeCrAl coated cladding under steady state operation and its mechanical stability under pellet-cladding interaction during a reactor startup. This work summarizes BISON 2-D radial-axially symmetric (R-Z) and radial-circumferentially symmetric (R-θ) simulation results by comparing 20, 50, and 100-µm FeCrAl coating thicknesses as well as FeCrAl cladding to traditional Zr-4 cladding under pellet-cladding interaction condition. The results of this study will provide a feasibility analysis for steady state and transient operational response of FeCrAl coated, Zircaloy cladding. Furthermore, this work will provide a fundamental basis for the assessment of the ability of coated cladding, as an ATF candidate, to maintain or exceed current steady state reactor operating expectations.

1. Introduction

The dominate fuel design in the current U.S light water reactor (LWR) fleet utilizes uranium dioxide (UO₂) clad in a zirconium alloy. Years of research accompanied by 60 years of reactor operational experience has steadily generated technological advancements as well as an extensive experience base of both material response and commercial and test reactor data, under steady state and transient conditions. This has lead current fuel technologies to reach the limit of their performance capabilities. The U.S. Department of Energy (DOE) along with the commercial nuclear energy industry has placed an emphasis on developing new technologies to reach the limit of their performance capabilities. The U.S. Department of Energy (DOE) along with the commercial nuclear energy industry has placed an emphasis on developing new technologies to reach the limit of their performance capabilities. The U.S. Department of Energy (DOE) along with the commercial nuclear energy industry has placed an emphasis on developing new technologies to reach the limit of their performance capabilities.

Carmack et al., 2013; Kiran Kumar et al., 2016; Stachowski et al., 2016). The events at the Fukushima Daiichi nuclear power generating station coupled with recent material advancements have provided a strong motivation for the industry and regulatory agencies to improve nuclear fuel performance, safety, and reliability by developing new fuel designs to address the inherent vulnerabilities of current fuel form during beyond design basis accidents, while maintaining current operational standards. To address these vulnerabilities, the U.S. DOE has invested in developing the next generation of nuclear fuel with the specific goal of improving fuel rod response while increasing its ability to tolerate a severe accident for a considerably longer period of time, as compared to conventional UO₂-Zircaloy fuel forms (Carmack et al., 2013; Kiran Kumar et al., 2016; Stachowski et al., 2016).

Accident tolerant fuels (ATFs) are designed, in essence, to increase
The PCI failure mechanism (Montgomery et al., 2013; Williamson et al., 2016) is often attributed to radial and axial temperature gradients and is believed to be an important factor in fuel pellet cracks forming in brittle ceramic pellets as a result of large radial temperature gradients and are believed to be an important factor in the PCI failure mechanism (Montgomery et al., 2013; Williamson et al., 2016). In order to demonstrate the improved fuel rod response, the advanced fuel form must reduce cladding oxidation reaction kinetics, minimize hydrogen generation rate, improve cladding thermo-mechanical properties, and reduce fission product release (Carmack et al., 2013; Kiran Kumar et al., 2016; Stachowski et al., 2016). While fission gas release is primarily controlled by the thermo-mechanical state of the fuel pellet, hydrogen generation and steam kinetics phenomena are primarily driven by the cladding material exposed to the coolant. When exposed to high temperature steam, the steam interaction with zirconium-based cladding alloys results in an exothermic oxidation reaction. Furthermore, the rapid oxidation of the cladding produces large amounts of hydrogen coupled with the exothermic oxidation reaction, can lead to an explosion such as the one witnessed at the Fukushima Daiichi Unit 3. In addition to the increased reaction kinetics, the oxidation and hydride formations severely deteriorates the clad structural integrity which could lead to a cladding breach and subsequent release of fission products. One of the primary goals for ATF concepts is to minimize high temperature steam-clad reaction kinetics and ensuing, hydrogen generation, by either replacing the Zircaloy cladding with a new material or by applying a coating on the outer surface of the cladding with enhanced corrosion resistance at high temperatures. One proposed concept applies an iron-chromium-aluminum alloy (FeCrAl) coating on the cladding outer surface to improve the high temperature corrosion resistance (Carmack et al., 2013; Kiran Kumar et al., 2016). Under steady state conditions, FeCrAl offers an improved high temperature corrosion resistance similar to other stainless steels used inside the reactor core. Furthermore, FeCrAl being a stiffer material than traditional cladding types, may assist in mitigating fuel failure that have plagued the industry, such as debris and fretting (Stachowski et al., 2016; Rebak et al. 2016; Cox, 1990). Under accident scenarios, FeCrAl has the ability to form both chromium oxide and aluminum oxide, depending on the temperature regime, dramatically improving the materials resistance to superheated steam, in turn reducing the hydrogen generation rate and maintaining a coolable geometry for longer periods of time (Stachowski et al., 2016; Rebak et al. 2016; Cox, 1990). However, the base material of FeCrAl being iron is less transparent to neutrons than zirconium alloys, having an impact on fuel cycle and electricity generation cost. Furthermore, the release of tritium into the coolant is of increased concern for iron-based alloys. However, the addition of a coating to the cladding outer surface will help alleviate the neutron penalty associated with FeCrAl alloys while improving the cladding mechanical response under normal operating conditions, specifically pellet-clad interaction (PCI).

PCI in LWR fuel is a coupled thermal-chemical-mechanical process that can lead to a breach in the cladding and subsequent release of radioactive fission products into the coolant under certain conditions of operating history, power changes, and fuel rod design characteristics (Lyons et al., 1963; Roberts and Gelhaus, 1979; Garzaroli et al., 1979; Gaston et al., 2009). Reactor operating restrictions, which limit power maneuvering, have been utilized to mitigate PCI, however, they restrain operational flexibility, leading to a loss of power generation. PCI failures generally occur in previously irradiated fuel subsequent to a rapid increase in local power after a period of time at a lower or reduced power level. Pellet Cladding Interaction Stress Corrosion Cracking (PCI-SCC) failure is driven by localized stress in the vicinity of a pellet crack, combined with the presence of aggressive chemical species, such as iodine and cesium, which induces stress-corrosion-cracking (SCC) in the cladding (Montgomery et al., 2013; Williamson et al., 2016). Radial fuel pellet cracks form in brittle ceramic pellets as a result of large radial temperature gradients and are believed to be an important factor in the PCI failure mechanism (Montgomery et al., 2013; Williamson et al., 2016). During a local power increase from a condition of reduced or closed pellet-cladding gap, pellet expansion produces high contact forces between the fuel pellet and the cladding, and during rapid thermal expansion of the pellet, the fuel cracks open further, inducing tangential shear forces on the cladding inner surface. These tangential shear forces are a function of the pellet-cladding gap residual contact pressure prior to a power maneuver, power level at gap closure, interfacial friction, and the maximum local power.

The addition of a coating changes the mechanics governing PCI, first, by adding an exterior coating foreign to the parent material, and secondly, by reducing the Zircaloy thickness in order to maintain consistent fuel rod geometry characteristics. An outer surface FeCrAl coating has vastly different material response than Zircaloy. Under PCI conditions, the coated cladding will be exposed to temperature changes and mechanical load induced by the fuel pellet expansion. Unlike a uniform single cladding material, the coated cladding will need to be treated as a composite, accounting for each constituent’s thermal expansion, elastic moduli, irradiation and thermal creep, irradiation hardening, irradiation growth and/or swelling, and plastic deformation models. In order to accomplish this complex analysis, a fuel performance code will need to be geometrically flexible as well as be able to handle complex material models, the coupling between each material model, and interaction between the two constituents in the system.

Fuel performance codes capable of modeling these failure mechanisms are multi-dimensional finite-element codes. The BISON fuel performance code provides unique material and physical behavior modeling capabilities as well as its finite-element versatility of spatial geometric representations, and for these reasons, BISON will be used to better understand these coated cladding concepts under PCI conditions. BISON is built upon the Multi-Physics Object-Oriented Simulation Environment (MOOSE) developed at Idaho National Laboratory (INL) (Capps et al., 2016). MOOSE is a parallel finite element computational system that uses a Jacobian-free, Newton-Krylov (JFNK) method to solve coupled systems of non-linear partial differential equations (Capps et al., 2016).

Using advanced fuel rod modeling capabilities in BISON, this study will provide a foundational analysis to understand the underlying changes in mechanical stability and provide fuel designers and engineers with a fundamental understanding of the performance of coated clad, as an ATF candidate.

2. PCI modeling approach

The PCI modeling approach used in this article has been described in detail in Ref (Montgomery et al., 2013; Williamson et al., 2016; Rashid et al., 1988) and closely follows the methodology developed in Reference (Groschel et al., 2002; Sunderland et al., 1999; Nesbit et al., 2009; Capps et al., 2016). The fuel rod analysis effort consists of two main steps, which together are used in order to determine the fuel rod characteristics following a base irradiation as well as the fuel rods mechanical response under PCI conditions.

Step 1 consists of a steady state R-Z depletion calculation for the first cycle of operation, which establishes the fuel rod steady state conditions: pellet-cladding gap, plenum pressure, and released fission gas, following the first cycle of operation. The results of this analysis provided input conditions for the local-effects analysis step. Additionally, this assessment consists of a full-length R-Z analysis of the second cycle startup immediately following the first cycle of operations. The purpose of this analysis is to identify the axial location in the fuel rod where the maximum cladding hoop stress occurs.

The second step is the initial local effects simulations which uses output parameters from the R-Z simulations in Step 1 defined by the location in Step 2. The data is utilized as input for the 2-D radial-circumferential (R-θ) or full 3-D geometric model corresponding to the axial location identified from the previous step. A pictorial description of this process can be seen in Fig. 1.
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