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Thermal analysis of the melting process in a nuclear fuel rod

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HIGHLIGHTS

- Melting of a nuclear fuel rod under the accident conditions is analyzed.
- Enthalpy formulation with explicit finite difference discretization is adopted.
- Temperature histories and phase change interface positions are determined.
- The heat transfer coefficient and heat generation rate are two predominant factors.
- Classical lumped model fails to give accurate results for high heat transfer coefficient.

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ABSTRACT

Melting of nuclear fuel rods is a crucial issue that must be addressed when conducting simulation analyses of severe accidents in nuclear reactors. The present work aims to develop a mathematical model and carry out numerical simulation of one-dimensional heat conduction with phase change in a nuclear fuel rod including two regions: uranium dioxide fuel pellet and zirconium alloy cladding. The temperature-dependent specific heats, thermal conductivities and densities are considered in the present investigation. The enthalpy formulation of the melting process in the nuclear rod during full reactor power operation is solved by using a finite-difference method. The thermal analysis is divided into four different phases: (1) transient phase before the cladding melting, (2) cladding melting phase, (3) transient phase before the fuel melting and (4) fuel melting phase. Then, the effect of heat transfer coefficient between coolant and fuel rod on the temperature histories of the fuel pellet and the cladding is investigated. Finally, the melting process of a nuclear rod with decay reactor power after shutdown is also simulated.

1. Introduction

Core meltdown, which may happen due to the inadequate cooling of the nuclear fuel rods, is a serious accident scenario in nuclear reactors, such as the Three Mile Island, the Chernobyl and the Fukushima nuclear accidents [1]. A core melt accident occurs when the heat generated by the nuclear reactor exceeds the heat removed by the reactor cooling systems to the point where at least one nuclear fuel element exceeds its melting point. In a pressurized water reactor (PWR) or boiling water reactor (BWR), the melting of fuel rods may release radioactive material into the cooling water and cause the zirconium cladding to react with water to generate explosive hydrogen.

Many analytical and numerical approaches have been proposed to obtain the transient heat conduction response of the nuclear fuel rod. Based on the assumption that the heat transfer resistance between the fuel and clad remains constant for both steady-state and transient periods, Chen et al. [2] developed a simple conduction model with phase change for the transient analysis of a reactor fuel rod based on average properties and lumped parameter techniques, where the analysis is subdivided into four phases: transient phase before the clad melting, clad melting phase, transient phase before the fuel melting and fuel melting phase. Ghiaasiaan et al. [3] obtained the solution of the governing equations for the radial heat conduction in the fuel pellet and cladding by the lumped capacitance method, the method of lines and the integral method, respectively. Regis et al. [4] analyzed the transient heat conduction in a nuclear fuel rod by an improved lumped parameter approach, where $H_{1,0}$ and $H_{0,0}$ Hermite approximations were applied for the averaged temperatures and heat fluxes. Su and Cotta [5] presented a higher order lumped parameter formulation for simplified LWR (light water reactor) thermohydraulic analysis that the two-sided corrected trapezoidal rule ($H_{1,1}$ approximation) was used in the

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Nomenclature

c_p	specific heat, J/(kg·K)
h	convective heat transfer coefficient, W/(m^2·K)
H	enthalpy, J/kg
k	thermal conductivity, W/(m·K)
L	latent heat of melting, J/kg
g	volumetric heat generation, W/m^3
r	space coordinate, m
R	radius, m
s	solid–liquid interface location
t	time, s
T	temperature, K
Greek letters	
ρ	density, kg/m^3
Subscripts	
c	cladding
f	fuel
i	node
l	liquid phase
m	melting phase
s	solid phase

averaged temperature integrals for the fuel and the cladding, and plain trapezoidal rule ($H_{0,0}$ approximation) was used in the averaged heat flux integrals. Pontedeiro et al. [6] proposed an improved lumped-differential formulation for one-dimensional transient heat conduction in a heat-generating cylinder with temperature-dependent thermophysical properties typical of high burn-up nuclear fuel rods. To eliminate the paradox of an infinite thermal wave speed, Espinosa-Paredes and Espinosa-Martínez [7] examined the applicability of a fuel rod mathematical model based on Non-Fourier transient heat conduction as constitutive law for the Light Water Reactors transient analysis (LWRs). Kudryashov et al. [8] studied the temperature distribution in the nuclear fuel rod of the reactor taking the influence of the high burn-up into account, and found the solution for the temperature distribution in the nuclear fuel rod in the analytical form for the stationary behaviour of the reactor. To overcome the limitation of the separation of variables method, Singh et al. [9] presented the finite integral transform method to solve the asymmetric heat conduction problem in a multilayer annulus with time-dependent boundary conditions and/or heat sources.

The melting behaviour of fuel rods during an accident is basically determined by solving the heat conduction equation [10]. The typical approaches to investigate the phase change problems include finite-difference (FD) scheme [11], heat balance integral method [12], moving mesh approach [13], variable iteration method [14], lumped parameter model [15], etc. For example, Voller and Cross [11] presented an explicit FD scheme for the enthalpy formulation of one-dimensional freezing problems, and also developed an implicit FD algorithm of the Stefan problems with internal heat generation and melting temperature defined by a mushy region, simultaneously. Cheung et al. [16] studied numerically the process of freezing and melting occurring in a heat-generating slab bounded by two semi-infinite cold walls, and solved the dimensionless governing equations using the method of collocation with Hermite splines as approximating functions and Gaussian quadrature points as the collocation points. Using a quasi-steady approximation, Jiji and Gaye [17] examined analytically one-dimensional solidification and melting of a slab with uniform volumetric energy generation, and applied the results to two examples: solidification of a nuclear material and melting of ice. With the perturbation technique, Yu et al. [18] studied the planar solidification with time-dependent heat generation in a semi-infinite plane, where the results of the proposed method were validated by the exact solution of the classical Stefan problem without heat generation. To assess the recriticality potential for a liquid metal fast breeder reactor (LMFBR) following a core disruptive accident, El-Genk and Cronenberg [19] studied the freezing phenomena of molten fuel on a cold structure, and used the successive approximation technique to obtain a solution to the non-linear freezing problem. The effects of heat generation, viscous heat dissipation, temperature-dependent thermophysical properties and a convective boundary condition at the solidification front have been incorporated into the an analytical formulation. Crepeau et al. [20] derived approximation governing equations of the interface location in one-dimensional PCM structure with internally generated heat in cylindrical, spherical, plane wall and semi-infinite geometries. Extending previous work, Crepeau et al. [21] investigated the solid–liquid phase change driven by volumetric energy generation in a vertical cylinder by a quasi-static, approximate analytical solution, and studied numerically the effect of convection within the liquid region. Crepeau and Siahpush [22] presented a quasi-static analytical solution of melting process in a cylinder with volumetric heat generation, which was valid for the Stefan number less than one.

This work aims at analyzing heat conduction for the accident scenarios in the core of a pressurized water reactor (PWR), specially the thermal behaviours of uranium dioxide pellet and zirconium alloy tubing under high temperature conditions that can cause the core to melt. Based on the enthalpy formulation, the governing equation is discretized by explicit FD methods with the forward and central differential schemes for time and space derivatives, respectively. The transient heat conduction of the fuel pellet with internal heat generation is considered, while there is no heat generation in the cladding. The temperature-varying thermophysical properties such as thermal conductivity, specific heat and density of fuel and cladding are adopted in the analysis.

The manuscript is organized as follows. The governing equation of one-dimensional melting of a nuclear fuel rod and the corresponding enthalpy formulation of the problem are presented in the next section. The FD schemes adopted to the enthalpy formulation is described in the following section. The melting process of the fuel rod during full power operation are then presented. Subsequently, the effect of heat transfer coefficient between coolant and fuel rod on the temperature histories of the fuel pellet and cladding is investigated. The melting process in the nuclear rod with decay heat power after shutdown is also simulated. Finally, concluding remarks and some perspectives are provided.

2. The mathematical formulation

2.1. Transient heat conduction

Consider the transient heat conduction in a nuclear fuel rod, which consists of heat-generating cylindrical pellets of uranium dioxide (UO_2) fuel sealed within a Zircaloy cladding tube, as shown in Fig. 1. A gap between the pellets and the cladding tube is filled with pure helium as a high thermal conductivity fill gas. We simplify the problem by assuming axisymmetry of temperature distribution and neglecting the axial heat conduction term and the spatial variation of the heat generation across the fuel rod. The temperature-dependent thermophysical properties are considered in the fuel and the cladding. Under the above assumptions, we have the following governing equations with appropriate boundary and