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Transient heat transfer during depressurization from supercritical pressure

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ABSTRACT

Fuel assemblies of supercritical water-cooled reactors may experience temporarily very high cladding temperatures when the operating pressure is reduced from supercritical to sub-critical conditions, if the cladding temperature has been hotter than the Leidenfrost temperature before reaching the critical pressure. This situation may cause film boiling or post-dryout conditions on the cladding surface, associated with a poor heat transfer, even if the critical heat flux has never been exceeded. As long as part of the fuel cladding will be wetted, the dry zone will slowly be quenched afterwards, as the quench front will remove heat from the overheated zone to cool it down under the Leidenfrost temperature, enabling rewetting again. The process has been modelled assuming a quasi-steady-state approach, which is using steady-state heat transfer correlations for supercritical, dry sub-critical and wetted sub-critical conditions, a steady-state enthalpy distribution in the fluid at any pressure, but transient heat conduction in the cladding.

1. Introduction

Several concepts of Supercritical Water Cooled Reactors (SCWR) have been studied in Japan, Europe, Canada and China within the last 10 to 20 years to explore the future potential of water cooled nuclear reactors, aiming at higher efficiencies and lower specific plant erection costs than current pressurized water reactors or boiling water reactors. Examples are the Super Light Water Reactor and the Super Fast Reactor concepts studied by Oka et al. [1], the High Performance Light Water Reactor documented by Schulenberg and Starflinger [2], or the Canadian SCWR outlined by Yetisir et al. [3]. For each of these concepts, the reactor core is designed with vertical fuel rods, inside which UO_2 pellets provide a heat source by nuclear fission like in conventional nuclear reactors. The pellets are encapsulated in thin walled tubes, called "claddings", which are grouped to fuel assemblies to ease handling. Different from these reactors, however, the coolant is assumed to be operated well above the critical pressure at around 25 MPa, with a core inlet temperature of ~280 °C and a core outlet temperature of 500 °C or more to improve the thermal efficiency. At this high core outlet temperature, the coolant becomes superheated steam, which can be supplied directly to a high pressure turbine without the need of a steam generator, which minimizes costs. The maximum linear power of the fuel rods, i.e. the fissile power per unit length of a fuel rod, is typically limited to 39 kW/m at an outer diameter of the fuel claddings of ~8 mm, which results in a surface heat flux of more than 1500 kW/m^2. As supercritical water is an excellent coolant, an average mass flux of ~1600 kg/m^2 s turned out to be sufficient to keep the peak cladding temperature below 650 °C [2]. Operation at sub-critical pressure, i.e. below 22.064 MPa, is not considered in these concepts except for start-up, shut down or accidental conditions. In such cases, however, the low critical heat flux at near critical pressure might cause a temporary boiling crisis with significantly higher cladding surface temperatures.

Recently, a test of four fuel rods of 8 mm outer diameter, heated by nuclear fission and cooled with supercritical water inside a pressure tube, has been proposed by Ruzickova et al. [4], which is being designed now by a European consortium in a joint project "SCWR-FQT". The pressure tube is intended to be installed inside a nuclear research reactor. It shall simulate the first heat up step of the core of the High Performance Light Water Reactor with its linear heat rate up to 39 kW/m. Even though this test facility with supercritical water is rather small, compared with a commercial nuclear reactor, it will still require reliable safety systems for accident management and for removal of the residual heat, which continues to heat the fuel rods after shut down of the research reactor. Raqué et al. [5] report about the planned safety systems and summarize the status of safety analyses. In this context, the question arises if any safety concern has to be expected if the

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Nomenclature

A	flow cross section
A_C	cladding or tube cross section
c_p	specific heat
C_{sf}	constant of the Rohsenow equation
d	outer cladding diameter or inner tube diameter
d_H	hydraulic diameter
F	correction factor
G	mass flux
g	acceleration of gravity
h	fluid enthalpy
k	heat transfer coefficient
n	number of fuel rods in the fuel bundle
Nu	Nusselt number
p	pressure
Pr	Prandtl number
q'	linear heat rate
q''	surface heat flux
q_{cr}	critical heat flux
Re	Reynolds number
s	distance from the quench front
t	time
T	temperature
U	velocity of the quench front
U^*	dimensionless velocity of quench front
x	steam mass fraction
z	vertical coordinate
α	void fraction
β	thermal expansion coefficient
δ	equivalent wall thickness
ε	ratio of heat transfer coefficients
κ	thermal diffusion coefficient
λ	thermal conductivity
μ	viscosity
θ	ratio of temperature differences far from the quench front
ρ	density
σ	surface tension
Δh_{LG}	enthalpy of vaporisation

Subscripts

a	actual
b	bulk
c	critical
dry	dry wall
f	film
G	vapour
hom	homogeneous
L	liquid
LF	Leidenfrost
sat	saturated
sc	supercritical
w	wall
wet	wetted wall

system pressure is reducing from supercritical to sub-critical conditions, e.g. as a consequence of a malfunction of pressure control or as a consequence of a small break in the closed coolant loop.

Heat transfer phenomena occurring during depressurization from supercritical to sub-critical pressure have also been studied in the past for coal-fired power plants. There, different from a reactor core, the coolant is heated up inside tubes and not between fuel rods like in a fuel assembly, but the heat transfer mechanism is similar. With the aim to optimise part load efficiency, coal-fired power plants with supercritical, once through boilers are usually operated with a sliding pressure such that the pressure decreases from full load conditions proportionally with decreasing load. For example, if the live steam pressure at full load is 25 MPa, transition to sub-critical pressure occurs at about 90% load. The heat flux in these boiler applications, however, is typically limited to <400 kW/m^2, at a mass flux of 2000 kg/m^{2}s or more, which is significantly lower than the heat flux envisaged for a SCWR, while the coolant mass flux is even higher. Therefore, we need to take a closer look on the physical phenomena to be expected during depressurization, to avoid overheating of the fuel claddings of a Supercritical Water Cooled Reactor during depressurization.

2. Analytical model

The heat transfer phenomena expected during a depressurization transient shall be discussed here with the help of a simple analytical model. Consider a vertical bundle of n fuel rods with outer diameter d and flow cross section A, as sketched in Fig. 1, which is heated such that it causes a heat flux $q''(z)$ on its wetted surface to an upward flow with mass flux G and pressure p. Under steady-state conditions, the heat balance yields the fluid enthalpy h, which increases with height z as

$$\frac{\partial h}{\partial z} = \frac{n\pi d}{GA} q'' \tag{1}$$

The local bulk enthalpy T_b (h,p) can be determined from this enthalpy using the steam table.

2.1. Heat transfer at supercritical pressure

Initially, the fluid shall be at supercritical pressure. Then the surface temperature can be determined e.g. with the correlation of Cheng et al. [6], which has the advantage of being explicit and does not need to be iterated like other correlations for this application:

$$Nu_b = 0.023\, Re_b^{0.8} Pr_b^{0.33} F \quad Re_b = \frac{G d_H}{\mu_b} \tag{2}$$
$$F = \min(F_1, F_2)$$
$$F_1 = 0.85 + 0.776(1000\pi_A)^{2.4} \quad \pi_A = \frac{\beta_b q''}{c_{p,b} G}$$
$$F_2 = \frac{0.48}{(1000\pi_{A,pc})^{1.55}} + 1.21\left(1 - \frac{\pi_A}{\pi_{A,pc}}\right)$$

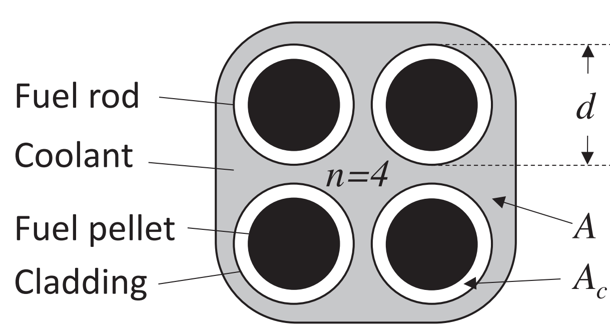

Fig. 1. Sketch of a bundle with four fuel rods.