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Instrumentation for experiments on a fuel element mock-up for the study of thermal hydraulics for loss of cooling or coolant scenarios in spent fuel pools

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ABSTRACT

Beside the nuclear reactor and its primary circuit the spent fuel pool is yet another safety-critical part in a nuclear power plant which has gained increasing focus after the Fukushima accident. Loss of coolant or enduring loss of cooling conditions would ultimately result in loss of cladding integrity at elevated temperatures with excessive release of fission products and hydrogen. To predict the available response time and to assess the efficacy of mitigating measures computer simulations can be employed. Their validity, however, needs to be proven by dedicated experiments at small scale but relevant thermal hydraulic conditions. For that purpose, the test facility ALADIN was designed, which enables conduction of experiments on a single BWR fuel element mock-up under loss of coolant and loss of cooling accident conditions. In this paper we introduce the facility and its instrumentation, with a focus on temperature sensors and a new thermal anemometry grid sensor for flow velocity measurement in one quadrant of the rod bundle with a resolution of one point per subchannel together with the affiliated calibration procedure for a potential application in superheated steam and air in a wide range of fluid temperatures.

1. Introduction

The spent fuel pool is an interim storage for spent fuel as well as fuel elements during revision periods in the reactor pressure vessel. Potential safety risks are the loss of cooling as a result of a longer persisting station black-out or the loss of coolant caused by a leakage. In such cases the integrity of fuel rods would be lost if a certain temperature limit is exceeded. This temperature limit depends on several factors, e.g. pool filling level and atmosphere, cladding material and individual properties of the cladding. The lowest temperature limit at which the integrity is just ensured and which should be used as a guide is at 565 °C. At this temperature time-dependent deformation (creeping) degrades cladding rigidity till failure after around 10 h (Smith, 1969; Nourbakhsh et al., 2002). A highly exothermic zirconium-air reaction occurs at cladding temperatures above 800–900 °C (Leistikow et al., 1975; Benjamin et al., 1979) and an exothermic zirconium-steam reaction above 1100 °C (Stuckert et al., 2011). The oxidation of zirconium by steam is accompanied by release of hydrogen gas presenting an additional hazard potential (Benjamin and McCloskey, 1980; Collins and Hubbard, 2001). This behavior has already been observed in the severe Three Mile Island accident at TMI-2, where a partial nuclear meltdown occurred as a result of a loss-of-coolant accident in the reactor pressure vessel (Kemeny, 1979).

Against this background and the lessons learned there is an increased need for simulation codes that can correctly predict the three-dimensional and transient heat transfer in the spent fuel pool and the resulting time course of the dry-out accident (NEA, 2008; Kaliatka et al., 2010). For the validation of the simulation codes reliable experimental data are required. So far experimental studies have been conducted in the US for air-cooled conditions on full-sized Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) assemblies to study fuel cladding ballooning and rapid zirconium oxidation (Lindgren and Durbin, 2013). Other experimental investigations were carried out on related aspects, but not originally with the focus on the investigation of spent fuel

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Nomenclature

A	area
A_ϑ, B_ϑ	temperature coefficient of resistance
BWR	boiling water reactor
C	thermal conduction constant
CVA	constant voltage anemometry
c_p	specific heat capacity
d	diameter
f_1	function of Pr
Gr	Grashof number
g	gravitational constant
K	curvature parameter
l	length
Nu	Nusselt number
n	amount
P_{el}	Joule heating power
Pr	Prandtl number $Pr = \mu c_p/\lambda$
$\dot{Q}_c$	convective heat flux
$\dot{Q}_t$	thermal conduction heat flux
R	electrical resistance
Ra	Rayleigh number
Re	Reynolds number $Re = \rho v l_{char}/\mu$
TAGS	thermal anemometry grid sensor
U	voltage
v	velocity

Greek

α	heat transfer coefficient
β	thermal expansion coefficient
$\Delta\vartheta$	overheating
μ	dynamic viscosity
ρ	density
ϑ	temperature

Subscripts

cal	calibration
char	characteristic
f	fluid
free	free convection
forced	forced convection
H	heating level
min	minimum
max	maximum
s	sensor overheated
T	temperature level
TC	thermocouple
w	wire
x	fluid composition

pools under accident conditions (Aksan et al., 1992). A more detailed description of past and present studies, both numerical and experimental, can be found in the status report of the Nuclear Energy Agency (Adorni et al., 2015).

Experimental investigations of the fuel load in an evaporating spent fuel pool or inside a reactor pressure vessel of a boiling water reactor has been conducted at the test facilities ADELA-I and ADELA-II at Technische Universität Dresden. In these projects preliminary simulations made with system thermal hydraulics codes were used to check their basic applicability. CFD codes have not been used so far. But this is considered as a crucial future need for a more accurate prediction of flow and heat transfer across multiple scales (Schuster, 2008; Schulz et al., 2014).

The main objectives of the German joint project SINABEL (**SI**cherheit **NA**sslager **B**renn**E**lement-**L**agerbecken) are the experimental investigation of the thermal hydraulics in a boiling water reactor fuel element and an improved CFD modelling. For this purpose a replica of a 10 × 10 rod BWR fuel element was constructed and is operated under spent fuel pool accident conditions. Special care was given to create boundary conditions for the single assembly close to those in a fuel rack with many assemblies. This is achieved by having heated bordering channels representing adjacent fuel elements in a simplified way. With the planned integral experiments thermal hydraulics effects within the fuel assembly as well as directly above can be studied.

In the following sections the investigated accident scenarios, the test facility design and its instrumentation will be presented. A stronger focus will be given to the special instrumentation, which had to be developed due to a lack of commercial products for this application.

2. Experiments and test facility

2.1. Investigated scenarios

For the investigation of both accident scenarios, the loss of coolant and loss of cooling respectively, two different initial conditions have to be considered. In the first case it is assumed that the coolant is absent due to damage of the pool integrity. The experiment will start in the dry state (Fig. 1, left). In contrast the loss of cooling accident experiment will start in the flooded state. The temporal progress of the flooding state depends on the decay heat or the heat input to the experimental system respectively. If the heat cannot be dissipated sufficiently via thermal transport processes the coolant will reach boiling temperature. Then the water level will decrease due to the evaporation of the coolant (Fig. 1, right). Parameters of interest in both scenarios are the surface temperatures of the rods and their temporal development as well as temperatures and flow conditions in the subchannels. This requires the determination of the fluid temperature and flow velocity in the subchannels of the rod bundle.

2.2. The test facility ALADIN

The ALADIN test facility is essentially a box-shaped enclosure containing a heavily instrumented BWR fuel assembly mock-up with some additional components to mimic conditions in a spent

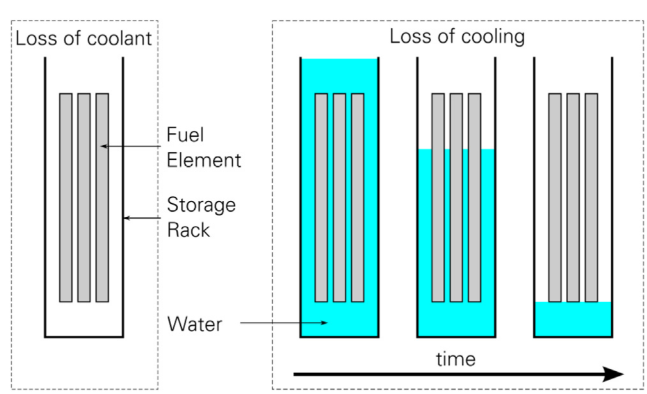

Fig. 1. Scheme of the states in a loss of coolant (left) and a loss of cooling (right) scenario for spent fuel pools.

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