Nuclear Engineering and Design 330 (2018) 480–487

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

# Experimental investigation of the thermal hydraulics of a spent fuel pool under loss of active heat removal conditions

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ARTICLE INFO

*Keywords:*
Spent fuel pool
Loss of cooling/loss of coolant
Accident conditions
Geysering
SINABEL
ALADIN

ABSTRACT

In case of a temporal undefined failure of the active cooling system of a spent fuel pool (SFP), the water inside the pool heats up depending on the decay heat of the stored spent fuel. In the worst case, this leads to a boil-off scenario with a water level decrease and, after appropriate time, to a progressive uncovery of the fuel assemblies (FA). The cooling of the uncovered structures is limited by the steam production in the lower still wetted part of the FA, radiative heat transfer, radial heat conduction and convection inside the FA and interaction with ambient air depending on the storage conditions. As a result, the removal of decay heat can be endangered and, by reaching certain cladding temperatures, the integrity of the cladding tubes can be compromised. The detailed heat and mass transfer mechanisms of the boil-off scenario are not yet sufficiently investigated – especially experimentally. Therefore, reliable conclusions about the development of the cladding temperatures cannot be made.

Against this background, the SINABEL project was launched in 2013 at TU Dresden for an improved understanding of the heat transport processes inside a SFP under accident conditions. Studies are performed both experimentally on the test facility ALADIN and theoretically by using computational fluid dynamics (CFD) calculations and integral codes for severe accidents. The test facility simulates a generic boiling water reactor (BWR) FA on almost original scale with regard to the boundary and ambient conditions within a SFP. Through an appropriate unique net of instrumentation, the distribution and development of radial and axial cladding temperatures and the development of the water level can be investigated in detail.

Experiments were carried out for various decay heats respectively rod powers and changing storage distances. The thermal hydraulics of the heat-up and the boil-off phase with the occurrence of subcooled boiling in the beginning followed by geysering-like instabilities are described in the present paper. The higher the heat load, the faster the heat-up and, as boiling temperature is reached, the lowering of the water level. The maximum limit of the water level and the influence by the radial heat conduction, convection and radiative heat transfer is shown and explained.

The experimental data and thermohydraulic findings provide the basis for the phenomenological understanding of this accident scenario and are used for the validation and improvement of numerical simulations.

## 1. Introduction

Although the safety of SFP has always been part of the scientific research, the accident in the nuclear power plant Fukushima drew attention to new research activities in this field. In addition, the safety requirements for SFP have changed over the time due to a denser storage and longer storage periods of the spent fuel. Despite great timeframes for counteractions e.g. in case of a failure of the active pool cooling systems that ensure the decay heat removal, one must be aware of the huge activity inventory[1] that is affected when no corrective actions are taken.

The investigations to date deal with three basic accident scenarios: The partial (1) or total (2) loss of coolant with the investigation of air-cooled FA and the loss of cooling (3), which leads to a heat-up of the water inventory and in the worst case to a boil-off with a water level decrease and an uncovering of the FA. In the 1980's Sandia National Laboratory and Brookhaven National Laboratory analyzed the spent fuel heat up in the case of a completely drained pool (second scenario)

*Abbreviations:* SFP, Spent fuel pool; FA, Fuel assembly; BWR, Boiling water reactor; CFD, Computational fluid dynamics; PSI, Paul Scherrer Institute; LOCA, Loss of coolant accident

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[1] Almost 100% of the total activity of a nuclear power plant can be stored in the SFP e.g. during revision.

https://doi.org/10.1016/j.nucengdes.2018.02.023
Received 24 May 2017; Received in revised form 13 February 2018; Accepted 15 February 2018
Available online 20 March 2018
0029-5493/

where FA are cooled by air with the computer codes SFUEL, SFUEL1W and the enhanced SHARP-Code (Sailor et al., 1987; Nourbakhsh et al., 2002). The third scenario was analyzed theoretically using thermal hydraulic codes like ATHLET, ATHLET-CD, MELCOR, COCOSYS and computational fluid dynamics (CFD) methodology (Kaliatka et al., 2010; Lin et al., 2012; NEA Committee on the Safety of Nuclear Installations, 2008). A comprehensive analysis of all three scenarios (boil-off, partial loss of coolant, total loss of coolant) was carried out by the Paul Scherrer Institute (PSI) with the computer code MELCOR (Jaeckel et al., 2014).

For the validation of the simulation codes, reliable experimental data are required. So far, a few experimental investigations on the decay heat removal of FA in evaporating SFP have been carried out. Rather, the studies refer to the case of an evaporating reactor pressure vessel respectively to a loss of coolant accident (LOCA) in a loop of the primary circuit with a water level decrease in the rector pressure vessel. The boundary conditions differ strongly in terms of significantly higher performance of the FA, smaller gap distances and pressure levels. Experiments conducted by the PSI on the NEPTUN test facility investigated a LOCA at low-pressure conditions. Reflood experiments should provide information on the heat transfer mechanisms between rod cladding and coolant and evaporation experiments were performed to investigate the decrease of the mixture level and the resulting rod cladding temperatures (Aksan et al., 1992). These reflood experiments were also carried out at the QUENCH test facility (Stuckert et al., 2014).

In order to obtain conservative results on the thermal hydraulic characteristics of a SFP, the conditions have to be adapted in order to draw correct conclusions in the event of a postulated failure. Studies at TU Dresden in collaboration with Vattenfall Europe Nuclear Energy investigated the boundary conditions of SFP. Experiments were carried out for air-cooled BWR FA and for BWR FA under the conditions of a boil-off scenario in SFP on two integral test facilities in original axial scale. Fig. 1 shows the simplified sectional view of both test facilities, called ADELA-I and ADELA-II. Both test facilities were composed of an inner channel and an outer channel, which are hydraulically connected to each other at the lower and upper end.

The test facility ADELA-I (2007–2010) simulated a part of a BWR FA with a 3 × 3 electrically heated rod bundle with a heated length of 3760 mm with one additional heater to reduce heat losses to the surrounding. The investigations gave a first approach for the analysis of the development of the cladding temperatures of FA in SFP under accident conditions (Schuster and Ohlmeyer, 2008). However, due to low instrumentation and large heat losses, the complex flow conditions could not be investigated. In the improved test facility ADELA-II, the investigation area was increased to a quarter of a FA with 8 additional electrically heated rods with the same heated length of 3760 mm. The results showed, that the radial heat flux has a significant influence on the temperature development of the rod cladding and which in turn makes it possible to conclude that the heat transport mechanisms between the gaps of the individual FA are decisive for the coolability of the rods. However, the influence of a change in the gap size could not be further investigated since a change in the system was not possible (Schulz et al., 2013).

For this purpose, the test facility ALADIN was built-up in 2016 within the joint project SINABEL that is funded by the German Federal Ministry of Education and Research (BMBF). The thermal hydraulic behavior of a BWR FA in a SFP under accident conditions is experimentally investigated on a full-size FA. In addition, the data is used for the validation of numerical simulations.

## 2. Test facility ALADIN

The test facility ALADIN adopts the geometric boundary conditions of BWR FA in nearly original scale taking the heat transfer mechanisms with the surrounding into consideration. The structure of the rod bundle, which is experimentally investigated, should not be too specific

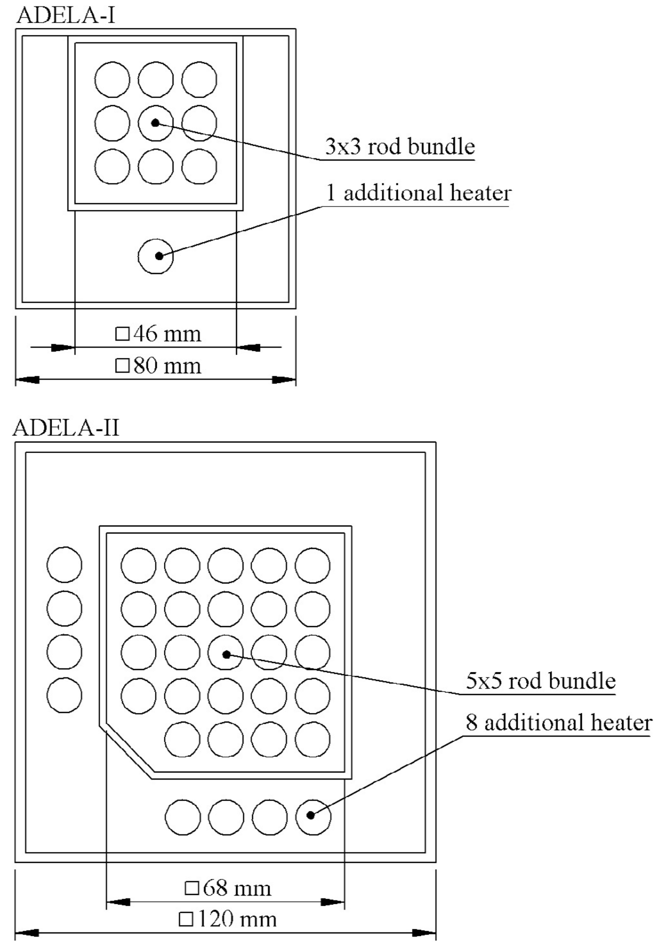


**Fig. 1.** Simplified sectional view of the test facilities ADELA-I and ADELA-II.

and rather represent a generic FA. The centered rod bundle consists of 96 electrically heated full-length rods with a diameter of 10 mm in a square arrangement, a lattice distance of 12.4 mm and a heated zone of 3600 mm. Generic spacers are used to keep the rods in distance to each other. The water channel in the middle of the rod bundle is used for instrumentation. An inner channel that simulates the FA box surrounds the rod bundle. Six additional channels with 44 heated rods simulate adjacent FA. The gap distance between the inner channel and the surrounding elements is variable between 5 and 50 mm (Fig. 2). All channels are hydraulically connected at the bottom and top, which can be seen in the simplified longitudinal view of the test facility in Fig. 3.

The axial power distribution of the electrically heated rods is in accordance to the characteristic power profile of a FA at the end of the core cycle under conservative assumptions (Table 1). Through 13 individually adjustable power supply units a radial power profile can be theoretically created. In recent experiments, a constant profile was chosen.

Most critical parameters for the safety assessment are the cladding temperatures of the rods and the relation to the water level. Temperatures of wall surfaces provide additional information about the thermal hydraulic phenomenology. Suitable instrumentation was selected and is presented in the following. In total 216 thermocouples were mounted on the claddings of the rods and on the channel surfaces on 12 different elevations in and outside of the test facility. These elevations or rather axial measurement planes have a difference of 400 mm (or between the two upper levels a distance of 350 mm) to each other. This axial division into measuring planes is given in Fig. 4. Thermocouples with a diameter of 0.5 mm are used inside the test facility for a minimal-intrusiveness and good temporal resolution and