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Mixed convection and stratification phenomena in a heavy liquid metal pool

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HIGHLIGHTS

- Results related to experiments reproducing PLOHS + LOF accident in CIRCE pool facility.
- Vertical thermal stratification in large HLM pool.
- Transition from forced to natural circulation in HLM pool under DHR conditions.
- Heat transfer coefficient measurement in HLM pin bundle.
- Nusselt numbers calculations and comparison with correlations.

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ABSTRACT

This work deals with an analysis of the first experimental series of tests performed to investigate mixed convection and stratification phenomena in CIRCE HLM large pool. In particular, the tests concern the transition from nominal flow to natural circulation regime, typical of decay heat removal (DHR) regime. To this purpose the CIRCE pool facility has been updated to host a suitable test section in order to reproduce the thermal-hydraulic behaviour of a HLM pool-type reactor. The test section basically consists of an electrical bundle (FPS) made up of 37 pins arranged in a hexagonal wrapped lattice with a pitch diameter ratio of 1.8. Along the FPS active length, three sections were instrumented to monitor the heat transfer coefficient along the bundle as well as the cladding temperatures at different ranks of the sub-channels. This paper reports the experimental data as well as a preliminary analysis and discussion of the results, focusing on the most relevant tests of the campaign, namely Test I (48 h) and Test II (97 h). Temperatures along three sections of the FPS and at inlet and outlet sections of the main components were reported and the Nusselt number in the FPS sub-channels was investigated together with the void fraction in the riser. Concerning the investigation of in-pool thermal stratification phenomena, the temperatures in the whole LBE pool were monitored at different elevations and radial locations. The analysis of experimental data obtained from Tests I and II underline the occurrence of thermal stratification phenomena in the region placed between the outlet sections of the HX and the DHR, respectively. After the transition to natural circulation, for both tests, this region with temperature stratification moves downwards starting at the exit section of the DHR-system, while upper and lower vessel regions show a uniform temperature.

1. Introduction

The European Union produces 30% of its electricity via nuclear fission in so-called second and third generation light water reactors (LWR) and, for some countries, the nuclear fission represents part of their sustainable energy mix (Sustainable Nuclear Energy Technology Platform, 2010). Safety and waste issues have to be considered and managed carefully at the international level. In particular, nuclear waste must be managed appropriately. Currently, the adapted approach is geological disposal, possibly preceded by used fuel reprocessing. This latter depends on fuel cycle choices and waste management policies of an individual country. In any case, the time scale involved in geological disposal exceeds that of the history of accumulated technological knowledge. As a result,

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ALFRED	advanced lead fast reactor European demonstrator
ADS	accelerator driven system
CFD	computational fluid dynamics
CIRCE	CIRColazione Eutettico
CR	research centre
DHR	decay heat removal
EFIT	European Facility for Industrial Transmutation
FP7	7th Framework Programme
FPS	fuel pin bundle simulator
EC	European Commission
ENEA	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
ESNII	European Strategic Nuclear Infrastructure Initiative
GEN-IV	GENeration Four
GIF	Generation Four International Forum
HLM	heavy liquid metal
HX	heat exchanger
IVCS	insulation volume cooling system
LBE	lead–bismuth eutectic alloy
LFR	lead cooled fast reactor
LMFR	liquid metal fast reactor
LWR	light water reactor
MHYRRA	multi-purpose hybrid research reactor for high-technology applications
Nu	Nusselt number
P&T	partitioning and transmutation
PLOF	protected loss of flow
PLOHS	protected loss of heat sink
Pe	Peclet number
Re	Reynolds number
SG	steam generator
SNE-TP	Sustainable Nuclear Energy-Technology Platform
SRIA	Strategic Research and Innovation Agenda
THINS	thermal-hydraulics of innovative nuclear systems
TC	thermocouple
XT-ADS	experimental accelerator driven system

geological disposal of nuclear waste does suffer from public acceptance problems.

In various studies, partitioning and transmutation (P&T) in critical and/or sub-critical fast spectrum transmuters has been identified as a way of reducing the volume and decay time of nuclear waste (Seventh Framework Programme FP7, 2014). This reduces the required monitoring period to technologically feasible and manageable time scales. Also in the framework of the GEN IV initiative this approach has already been put forward.

On a European level a collaborative effort supported by the European Commission (EC) and European research institutes and industries was started to bring advanced fuel cycles and the P&T strategy together, in order to investigate its economic and technical feasibility (Sustainable Nuclear Energy Technology Platform, 2010). The exploratory research done in the field and the launch of the Sustainable Nuclear Energy Technology Platform (SNE-TP) in 2007 lead to a joined effort from the European nuclear fission research community to issue a Strategic Research and Innovation Agenda (SRIA) that describes the roadmap towards sustainable nuclear fission energy.

SNE-TP community identifies sodium fast reactor technology as the reference but also highlights the need for the development of an alternative track with lead or gas cooling. In addition, the need for R&D activities in support of accelerator driven systems (ADS) was stressed to allow the demonstration of ADS technology by the construction of the first ADS Demo facility (MYRRHA, De Bruyn et al., 2007).

Regarding alternative fast reactor technologies as described in the SRIA, lead cooled fast reactor (LFR) systems are very promising in meeting the Gen IV requirements in terms of sustainability, economics, safety and reliability and proliferation resistance & physical protection. This assessment is based on inherent properties of the reactor coolant and on basic design choices.

In the European Strategic Nuclear Infrastructure Initiative (ESNII) implementation plan, the roadmap for the development, design construction and operation of a lead cooled fast reactor is put out. The conceptual design of a LFR demonstrator (ALFRED) is foreseen (Alemberti et al., 2014).

Since the lead-cooled fast reactor (LFR) has been conceptualized in the frame of GEN IV International Forum (GIF), ENEA is strongly involved on the HLM technology development.

Currently ENEA has implemented large competencies and capabilities in the field of HLM thermal-hydraulics, coolant technology, material for high temperature applications, corrosion and material protection, heat transfer and removal, component development and testing, remote maintenance, procedure definition and coolant handling (Foletti et al., 2006; Cinotti et al., 2012).

In this frame the CIRCE pool facility (CIRCulation Eutectic) was refurbished to host a suitable test section able to simulate the thermal-hydraulic behaviour of the primary system in a HLM cooled pool reactor. In particular, a fuel pin bundle simulator (FPS) was installed in the CIRCE pool (Bandini et al., 2001; Tarantino et al., 2011). It was designed with a thermal power of about 1 MW and a linear power up to 25 kW/m, relevant values for a LMFR.

The aim of this experimental campaign performed in CIRCE facility arranged with the integral circulation experiment (ICE) configuration is to characterize the phenomena of mixed convection and stratification in a liquid metal pool in a safety relevant situation.

Most of the works available in literature, concerning natural circulation phenomena of interest in the nuclear field, deal with results obtained using water or sodium as working fluids (Ishitori et al., 1987; Watanabe et al., 1994). Furthermore, most of them neglect the thermal stratification that is instead considered one of the most important topics in the study of Generation IV reactors for increasing reactor safety and its structural integrity. Because of an accidental scenario, the reactor is scrammed, and assuming the total loss of the pumping system, the coolant flow rate reduces and large temperature variation takes place causing thermal stratification phenomena inside the pool. A stiff vertical temperature gradient may induce significant thermal loads on the structure in addiction to existing mechanical loads. Moreover, due to the instability (with respect to the position) of the stratification interface low frequency oscillations with large amplitude are generated. Since the thermal conductivity of HLM is 10–100 times higher than that of water (for lead at 450 °C the thermal conductivity is about 17 W/m K) temperature fluctuations are transmitted with low attenuation to the structure, leading to thermal cycle fatigue on the surface of the structure materials.

To this end, the transition from nominal flow full power conditions to natural circulation decay heat removal conditions was explored, investigating mixing and stratification in large pool. In order to investigate pool thermal-hydraulics and provide experimental data for the validation of CFD models, the on-set and stabilization of the DHR flow path was monitored by means of a suitable instrumentation. Several thermocouples were used in the 3D domain to map the thermal stratification during the transient.

Due to the integral nature of the facility, the tests will also be valuable for the verification of the system codes in