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Numerical simulations of an innovative decay heat removal systems

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Abstract The general interest in innovative forms of nuclear energy from fission leads many industrial companies and research institutions to develop ambitious projects to design an innovative reactor fleet. Often such new designs are based on innovative reactor coolants and imply as a consequence a complete change of the technology know how. Among the main characteristics of such developments, safety demonstration plays a central role. Safety systems of such newly developed designs must meet different requirements respect to those designed for reactors currently in operation. One good example is the case of the lead fast reactor, where primary coolant freezing must be considered. The paper reports an excursus of what has been done over the past few years on ALFRED reactor decay heat removal system, reporting the history of the safety system from the preliminary configuration up to the present, together with an innovative idea of a passive system to delay lead freezing. Finally, ongoing activities for the experimental qualification of such safety system are presented, together with support and pretest calculations.

1 Introduction

In the field of nuclear innovation for fission reactor technology, many private companies and national research centers are strongly engaged in research programs for the development of generation IV reactors. With reference to the European context for example, Ansaldo Nucleare is currently developing a lead fast reactor named ALFRED [1, 2], SCKCEN is leading the MYRRHA project [3, 4], CEA research center is developing ASTRID sodium reactor [5] while V4G4 Center for Excellence is pursuing the design of ALLEGRO gas fast reactor [6]. At the International level, a cooperative international endeavor, named Generation IV International Forum (GIF), has been set-up to carry out the research and development needed to *establish the feasibility and performance capabilities of the next generation nuclear energy systems* [7]. The main reason for such a huge interest lies in the capability of these reactor concepts to guarantee a sustainable source for future energy demand at an acceptable cost for the population while ensuring high security and safety standards [7]. Currently, the research is focused on several lines of development, including reactors cooled by heavy liquid metals such as lead [8]. Lead has many peculiarities that make it an ideal candidate as a coolant, including its good thermal and neutron properties that

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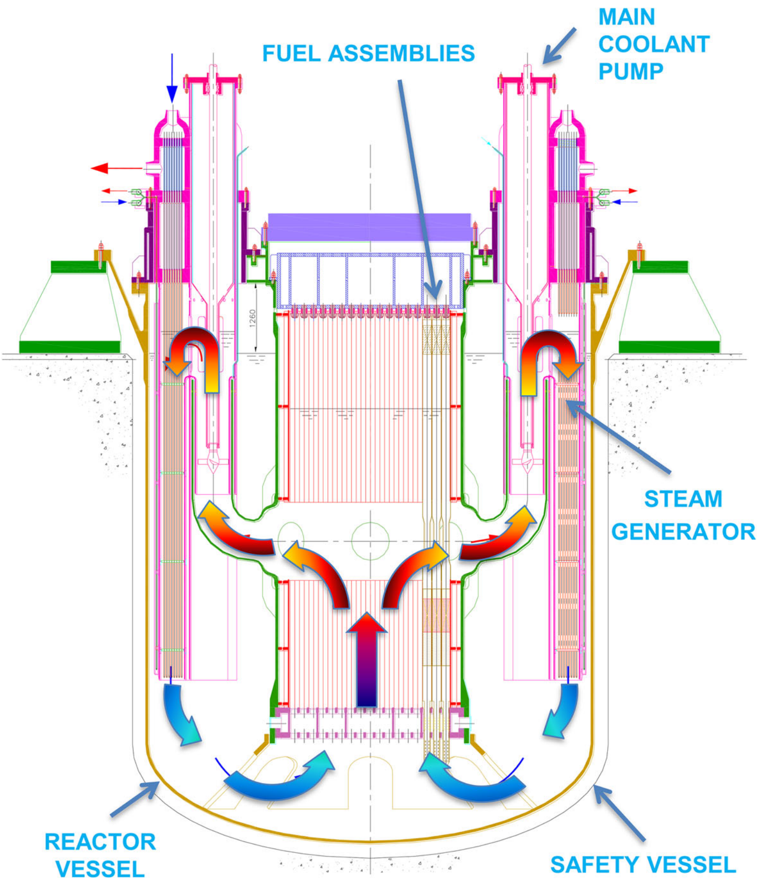


Fig. 1 Cutaway of ALFRED as from LEADER project (axial)

allow it to cool the core efficiently and guarantee a fast neutron spectrum. The technology is safe because it allows to cool the core during accidental conditions by means of passive safety systems in which lead transfers heat by natural circulation. Clearly, opacity and corrosive aggressiveness to steels pose challenges to the fast technology deployment. For more than a decade, Ansaldo Nucleare has supported the development of lead reactor technologies through internal research and development campaigns and also by taking part in international research projects [9, 10], working on design solutions for the primary, secondary and safety systems. The paper aims to provide a historical account of some developments that have been carried out in the field of safety systems for lead reactors, with particular reference to the ALFRED reactor, starting from the EU-funded project LEADER in FP7 [11]. The first conceptual designs are presented for the decay heat removal systems (DHR), the issues encountered and the solutions that have been identified. Finally, the last configuration of the system is presented together with the numerical activities to support the experimental campaigns currently in place for the qualification of the safety system.

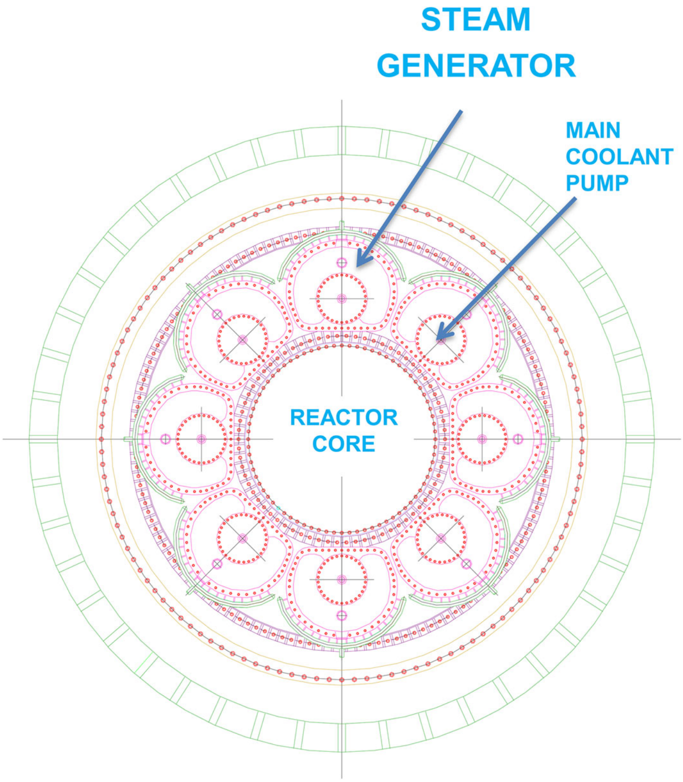


Fig. 2 Cutaway of ALFRED as from LEADER project (top)

2 Summary of ALFRED DHR development

The LEADER project [11] has been the starting point in the design of the ALFRED reactor as a demonstrator unit. A section of the system configuration for the primary system—of the pool type—is shown in Figs. 1 and 2.

The primary system provides for the central region occupied by the core. Leaving the core, the hot lead passes through 8 pump channels [12] from where it is channeled into the 8 bayonet-type steam generators (SG) [13, 14] where it is cooled to enter the core again. Inside the SG, on secondary side, the feedwater (FW) enters a bayonet tube bundle to exit in the form of strongly superheated steam (450 °C and 180 bar) conveyed to the turbine through the steam line (SL).

2.1 ALFRED DHR during LEADER European project

The DHR systems are connected to the SG through the feedwater (FW) and the steam line (SL) as shown in Fig. 3. Eight loops are placed in parallel and divided into two trains, respectively, called DHR1 and DHR2. Three loops are sufficient to correctly remove the decay heat. They consist of heat exchangers immersed in a pool in environmental conditions (Isolation Condenser, IC), with the tubes welded to spherical manifolds. In the event of an accidental condition, the reactor pumps trip and SCRAM are followed by the isolation of

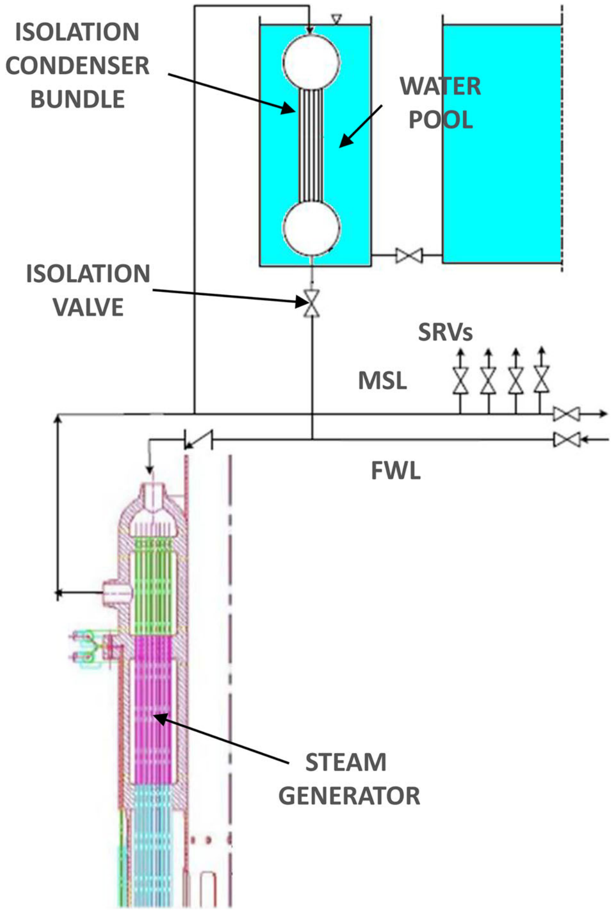


Fig. 3 ALFRED DHR during LEADER

the FW and SL, followed by the passive opening of the valve at the bottom of the IC. From this moment, lead transfers heat from the core to the SG by means of natural circulation, water in the DHR transfers heat from the SG to the IC by means of boiling and condensation phenomena while the water in the pool transfers heat to the external environment by means of mass transfer (boiling). Pressure in the DHR is controlled by the safety relief valves (SRV) which open in case of overpressure and close when pressure decreases below the set point.

The design requires the actuation of few valves and it is considered passive of category D according to the IAEA definition [15]. The performance verification of the DHR was tested adopting a numerical model used with the system code RELAP5-3D© [16] which provides for the simulation of the primary system, the DHRs and part of the secondary circuit. The model was used against two accidental scenarios to evaluate the operating conditions of minimum and maximum performance, respectively. Each transient postulates the loss of heat sinks (LOHS) for normal operation and the reactor SCRAM, starting from normal operation. Then, in one case 3 DHR loops out of 4 are actuated (one is supposed to be failed) and the primary pumps trip (minimum performance), while in a second phase 4 DHR loops are actuated out of 4 and the primary pumps are left on (maximum performance). Figure 4

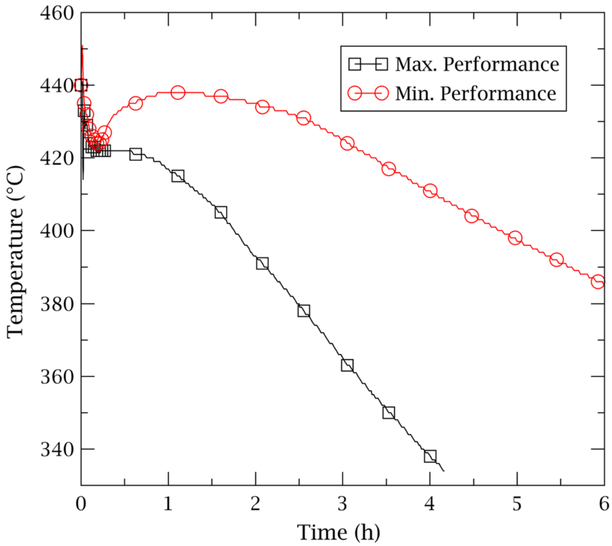
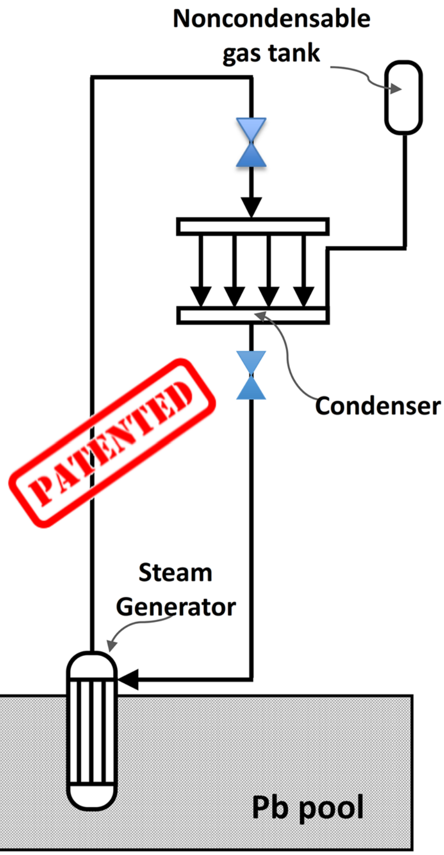


Fig. 4 Average lead temperature during the accident

Fig. 5 Updated DHR system



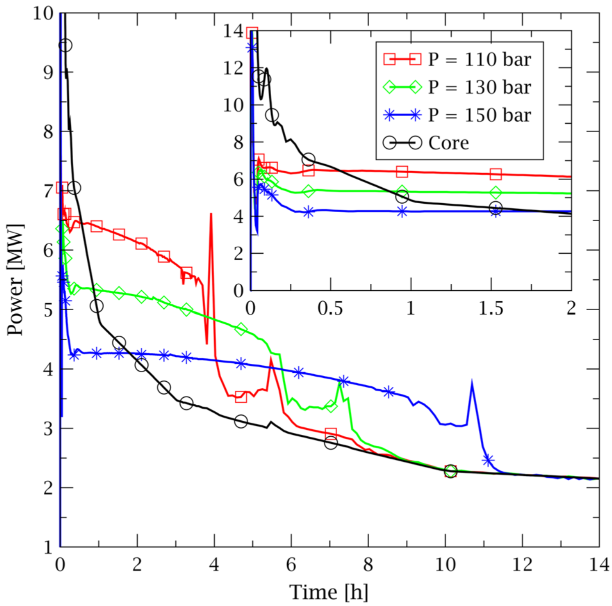


Fig. 6 Power balance [19]

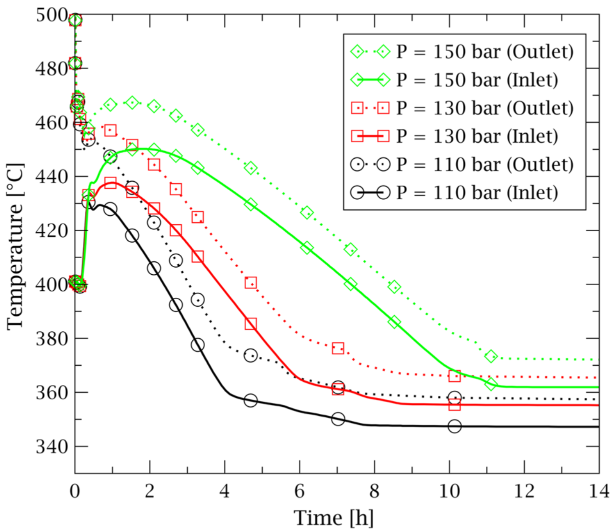


Fig. 7 Lead temperature at core inlet (straight) and outlet (dashed) [19]

shows the average lead temperature during the transient. Both cases result in acceptable maximum temperatures (in fact, maximum lead temperatures are below normal operational values demonstrating the acceptability of the solution). As far as lead freezing is concerned, maximum grace time (the time after which manual operations are needed to reduce power extraction from DHR) varies between 5 and 8 h.

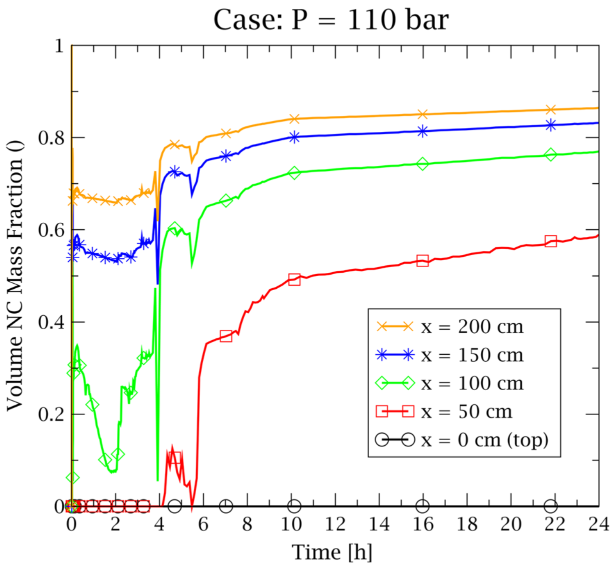


Fig. 8 Non-condensable gas concentration in IC bundle [19]

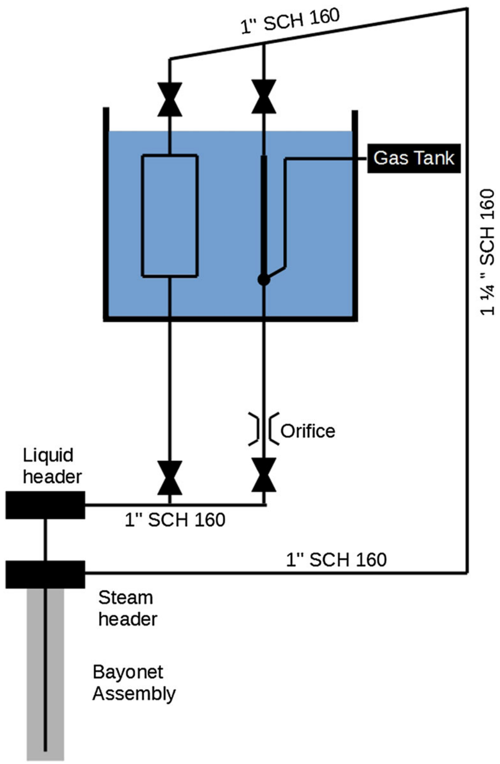
2.2 A passively controlled DHR system

When the LEADER project was completed, a design review was carried out which led to the identification of some margins for improvement, in particular on the DHR system. The critical points of the design were essentially 2:

1. The operation of the relief valves during the accidental transient poses reliability issues, as well as a basic question from the point of view of safety. If on the one hand the safety valves must be designed to fail open, in order to avoid catastrophic failure of the system in the event of an overpressure, the operation of these valves during an incidental transient would require them to fail in the closed position, to not lose the system's water inventory with consequent depressurization.
2. The problems concerning primary coolant freezing impose manual operations on the system which, although simple, must be carried out after few hours from the accident beginning. The nuclear accidents of history have taught us that in the first place it is not obvious that it is possible to carry out actions in the regions in the proximity of the containment during the first phases of the transient, and that the probability of operator error decreases as one moves far in time from the initiating event.

To resolve these critical issues, some system requirements have been changed. To prevent an overpressure of the system in the initial phases of the transient an additional volume is needed to discharge the water mass contained in the SG. Further, to delay the onset of problems due to freezing, a research and development activity was carried out to develop the idea of a system capable of passively modulating the power removed, without operator intervention and using simple physical principles. The idea that was proposed was to use non-condensable gases: From the theory [17] it is known how the condensation efficiency of a fluid in the presence of non-condensable gases proportionally degrades with gas concentration. To ensure that the mass transfer of the gases works as required and satisfy also the first requirement of an additional volume, the system has been calibrated considering its total

Fig. 9 SIRIO facility



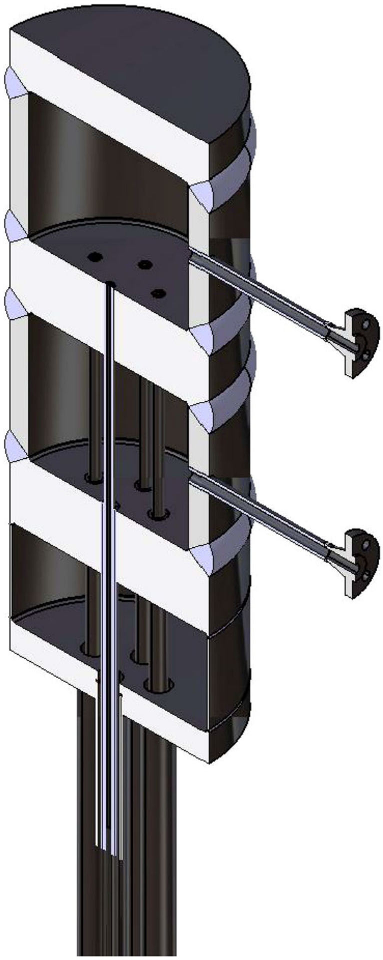
volume, the water inventory trapped between the FW and the SL isolation valves and the non-condensable gas mass required. This is brought to the system configuration in Fig. 5.

Compared to the previous configuration, the changes are as follows:

1. The IC now has two valves, one at the inlet and one in the injection line. Between the valves, a non-condensable gas under pressure (nitrogen) is interposed.
2. A tank is connected to the lower manifold of the IC.

During normal operation of the reactor, the DHR valves are closed and keep the gas confined. In the event of an accident, the FW and SL isolation valves close. Once an overpressure set point is reached, the DHR admission valve opens, the steam enters the IC and pushes the non-condensable gases into the tank. After a delay, the DHR return valve opens and the water can return to the SG. The system continues its operation in this situation until the power removed from the reactor balances the evacuated power toward the IC water pool. When the reactor power drops due to the decay heat exponential form, the power mismatch causes the DHR system to reduce its pressure. At that moment, non-condensable gases reenter the system and are inserted in proportional quantity into the IC tube bundle to counteract the pressure reduction. This has the effect of decreasing the efficiency of the IC and bringing the power produced and power removed back to balance. The innovative idea was presented and accepted as a patent [18]. The principle of the safety system was tested using the numerical model of ALFRED previously developed in the LEADER project, to which changes were made to the safety system. A sensitivity study was also carried out to assess the impact of the initial nitrogen pressure (between 110 and 150 bar) [19]. The results showed that the operating principle fully meets the demands of the design. Figures 6, 7 and 8 show the power

Fig. 10 SIRIO SG



trend produced by the core and removed from the DHR, the temperature trend of the lead (core input and output) and the concentration pattern of the non-condensable gases on the length of the IC bundle.

2.3 SIRIO facility

The qualification of any innovation for the installation on a nuclear plant, in particular those related to safety, must pass through a series of experimental activities that support its functionality and intended goals. To test the innovative safety system for ALFRED reactor, the SIRIO project, co-financed with the Italian Ministry of Economic Development and part of a future European project, was proposed with the aim of building and using an experimental facility that gives physical evidence of the expected phenomena as well as to serve as a unique tool for simulation tools validation. Currently, the facility is in the procurement and construction phase, and experimental tests are scheduled for the beginning of 2020. Figure 9 shows a simplified diagram of the SIRIO facility. The system includes the electrically heated bayonet SG and the IC inside its own pool, connected to the gas tank. The facility is scaled

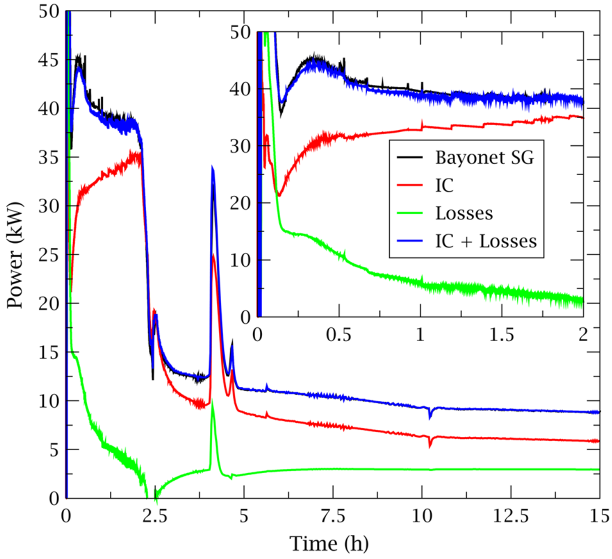


Fig. 11 Power balance

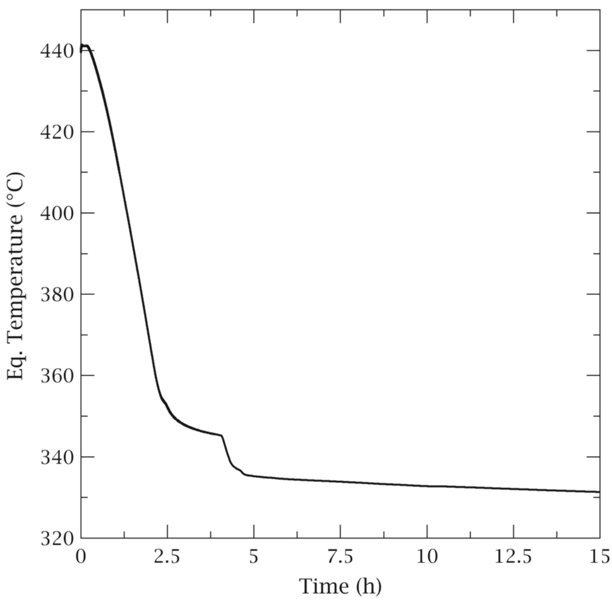


Fig. 12 Equivalent lead temperature

with the method of constant power density, full height and pressure, and the scaling ratio is about 1:47. The steady state conditions of the system are simulated by means of a line in parallel to the IC with a pool heat exchanger, to avoid the introduction of feed pumps. Figure 10 shows a detail of the bayonet steam generator.

To ensure that the experimental facility responds consistently with what was expected, a Relap5-3D© model of the facility was developed and some pretest calculations were carried

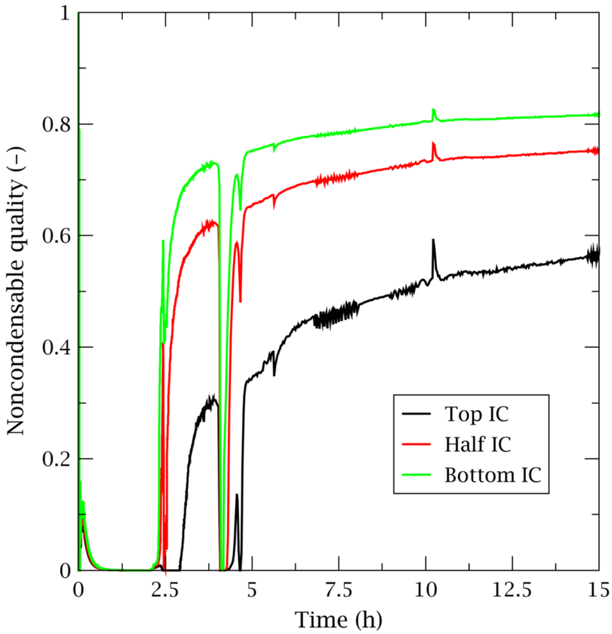


Fig. 13 Non-condensable quality in IC

out [20]. Particular attention has been devoted to the modeling of heat losses of the facility that greatly affect its operation, given the small size compared to the real system. Figure 11 shows the heat duty in the different components of the DHR, together with heat losses. Coupled with Fig. 12 (equivalent lead temperature based on SIRIO heat balance) and Fig. 13 (concentration of non-condensable gases calculated in certain points of the IC), it clearly shows the impact of non-condensable gases, and the IC removed power is able to follow the heat input from the SG. Overall, the most important phenomena affecting the DHR are also expected on the scaled experimental facility. As the power source from scaled decay heat curve imposed on the bayonet SG decreases, the concentration of non-condensable gases increases over time in the IC to continuously counterbalancing the decrease in heat source by decreasing IC efficiency.

3 Conclusions

The paper presents the development of the safety system for the decay heat removal of ALFRED reactor. The first version of the system, developed during international collaborations within the European LEADER project, was judged sufficient from the point of view of the power requirements necessary to keep the reactor below temperatures challenging the integrity of structural materials, but highlighted margins for improvement regarding the delay of freezing of the primary coolant. With this in mind, Ansaldo Nucleare developed an innovative system capable of passively control the power removed from the safety system with the aid of non-condensable gases placed in strategic regions of the system. The preliminary calculations carried out with the aid of the RELAP5-3D code have confirmed that the modifications on the system allow to significantly delay the freezing of the primary coolant, increasing the plant's grace time. In support of numerical activities, a research project partially funded by the Italian Ministry of Economic Development (SIRIO) was launched to build an experimental facility scaled on the safety system so as to gain experimental evidence of the operation. The facility will also be subject to future tests as part of a forthcoming European project already funded in the frame of H2020 research program for application on advanced and existing nuclear reactors.

References

1. M. Frignani et al., in *Proceedings of SESAME International Workshop* (Petten, 2019)
2. M. Frignani et al., *to be published in Proceedings of ICONE 2019 Congress* (Juan-les-pins, 2019)
3. D. De Bruyn et al., in *Proceedings of ICAPP 2018* (Charlotte NC, 2018), p. 1074–1079
4. D. De Bruyn et al., in *Proceedings of ICAPP 2018* (Charlotte NC, 2018), p. 1066–1073
5. T. Beck et al., Nucl. Eng. Des. **315**, 51–60 (2017)
6. B. Kvidza et al., Nucl. Eng. Des. **345**(15), 47–61 (2019)
7. J. Kelly et al., Prog. Nucl. Energy **77**, 240–246 (2014)
8. A. Alemberti et al., Prog. Nucl. Energy **77**, 300–307 (2014)
9. M. Grattarola, ELSY Publishable final activity report, 2010, Available at: https://cordis.europa.eu/project/rcn/80053/factsheet/en. Accessed 12 Jan 2020
10. M. Schyns, MAXSIMA project presentation, 2013, Available at: https://maxsima.sckcen.be/en/Deliverables. Accessed 12 Jan 2020
11. A. Alemberti, Final Report Summary – LEADER, 2013, Available at: https://cordis.europa.eu/project/rcn/96603/reporting/it. Accessed 12 Jan 2020
12. W. Borreani et al., Energies **10**(12), 10 (2017)
13. M. Caramello et al., in *Proceedings of ICAPP 2015*, Nice
14. L. Damiani et al., Nucl. Eng. Des. **265**, 154–163 (2013)
15. IAEA, IAEA Safety Glossary, Terminology used in Nuclear Safety and Radiation Protection, 2016 revision, Vienna, June 2016
16. RELAP5-3D code development team, 2012. "RELAP5-3D Code Manual", Idaho National Laboratory
17. W.J. Minkowycz, E.M. Sparrow, Int. J. Heat Mass Transf. **9**, 1125–1144 (1966)
18. WO2015059672A1, 2015. Heat exchange system and method with passive regulation of the amount of heat removed
19. M. Caramello et al., Prog. Nucl. Energy. **99**, 127–139 (2017)
20. Marinari et Al., in *Proceedings of ICONE-26* (London, 2018)