Id.: EN39

ANALYSIS OF NITROGEN INTRUSION WITH TRACE5. APPLICATION TO AN SMALL BREAK LOCA

María Lorduy-Alós, Sergio Gallardo, Tomás Ten-Gallardo, Gumersindo Verdú

Universitat Politècnica de València, Camí de Vera s/n, València, Spain.
sergalbe@iqn.upv.es;

Keywords: Nitrogen intrusion, TRACE5, Large Scale Test Facility

ABSTRACT

After the injection of borated water during accidental sequences, the intrusion of nitrogen gas from the accumulators once emptied can degrade heat transfer mechanisms and thus hinder core cooling. The non-condensable gas that enters the cold legs transfers and accumulates in the U-tubes of the steam generators and their outlet plenums. This can cause flow stagnation and even prevent reflux cooling. Furthermore, a temporary and fluctuating increase in primary pressure may occur, disabling the low-pressure injection system pumps to restore the inventory. In this scenario, it is essential to confirm the effectiveness of the accident management measures under the influence of the non-condensable gas. An additional problem is the oxidation of the fuel cladding in the presence of nitrogen if the core temperature increases sufficiently. To date, several experimental studies have been conducted within the framework of OECD/NEA programs at experimental facilities such as LSTF, PKL, or ATLAS (all related to PWR-type designs) where the issue of non-condensable gas intrusion have been studied, but always as a secondary objective of the analyzed sequences. In this work, three SBLOCA in the reactor pressure vessel of the LSTF facility are analyzed to determine the effect of nitrogen intrusion in the primary system. For calculations, the thermal-hydraulic code TRACE5 patch 7 is used. Results include the evolution of the main thermal-hydraulic parameters (pressures, mass flow rates and temperatures) and sensitivity cases for different scenarios of nitrogen intrusion.

1. INTRODUCTION

In typical Pressurized Water Reactors (PWR), nitrogen and other non-condensable gases (NCGs) can enter the Reactor Coolant System (RCS) during various transients, including Loss-Of-Coolant Accidents (LOCA), each with distinct implications for reactor safety and performance. According to Karl, 2015 [1], nitrogen can enter the RCS from several sources: When RCS pressure drops, NCGs dissolved normally in the coolant can be released, as well as NCGs from the non-deaerated coolant of the ECCS or dissolved nitrogen in water from the accumulators that is injected during a LOCA scenario. Failure of the Accumulator (ACC) isolation valves can result in direct entry of nitrogen gas into the RCS. Furthermore, reverse flow at the break in a LBLOCA scenario may occur if primary pressure drops under containment pressure.
As a result of nitrogen build-up in the system, especially in the steam generator tubes, the efficiency of heat transfer reduces, leading to potential overheating and pressure stagnation. Degradation of heat transfer at SG and blocking of flow paths inhibit reflux condensation cooling. Over the past decades, several experimental programs at integral and separated-effect test facilities have included experiments aimed at studying these issues. Among others, BETHSY, PACTEL, PKL, and LSTF integral test facilities have included experiments in their programs to provide data for understanding how NCGs can impact reactor safety during various accidental scenarios [2-6]. Regarding Separate Effect Facilities (SEF), some experiments analyzed the influence of nitrogen on reflux condensation [7, 8], on laminar film condensation in vertical tubes [9] and on steam-air condensation on inclined surfaces [10, 11].

In this work, it is analyzed the tests SB-PV-07 [12], SB-PV-09 [13] and SB-PV-10 [14] of the NEA/OECD ROSA/LSTF programs, all three with nitrogen intrusion from ACCs. For this purpose, the thermal hydraulic code TRACE5 patch 7 is used. The aim of this work is assessing the capability of the thermal hydraulic code to reproduce the physical phenomena associated to the nitrogen intrusion in the primary system of LSTF under SBLOCA scenarios.

2. MATERIALS AND METHODS

2.1. LSTF facility

The LSTF experimental facility [15] is designed to simulate the second unit of the Japanese Tsuruga nuclear power plant (Tsuruga-II). This unit is a four-loop PWR, Westinghouse design, with a power rating of 3423 MW thermal (1100 MWe). This facility was designed by the Japan Atomic Energy Research Institute (JAERI), now the Japan Atomic Energy Agency (JAEA), as part of the ROSA-IV (Rig of Safety Assessment-IV) program in 1985. The installation has a Full-Height Full-Pressure configuration, preserving the heights and operating pressure characteristic of the reference plant. Regarding volumetric scales, two types of scales are available depending on the component of the installation. For the hot and cold legs, as the four original loops have been collapsed in two, a volumetric scaling of 1/24 is presented, while for the rest of the components a volumetric scaling of 1/48 is used. The primary system consists of the pressure vessel, two symmetrical cooling loops, a pressurizer and an ECCS. Each of the cooling loops is composed of a hot leg, a U-tube bundle, a loop seal, a centrifugal pump, and a cold leg. The power generated in the core is limited to 10 MW (14% of the scaled full power plant) and this is generated by 1008 heated rods. The secondary circuit consists of two steam generators, their corresponding valve trains and auxiliary feed pumps.

2.2. Experimental conditions

To study the effects of NCGs on the system, the SB-PV-07, SB-PV-09 and SB-PV-10 tests have been selected. Tab. 1 lists the sequence of major events and signals of the three experiments. The experiments reproduce a loss-of-coolant accident caused by a small break in the vessel. The break in the SB-PV-07 is located at the upper head and has an area equivalent to 1% of the cold leg, 10.3 mm inner diameter. The break in the SB-PV-09 test is also at the upper head and it is 1.9%, approximately 13.8 mm inner diameter. The break unit in SB-PV-10 is placed in the lower plenum and has an area equivalent to 0.1% of the cold leg, 3.2 mm inner diameter.

Tab. 1. Sequence of major events and signals in the experiments.

Event	Signal		
	SB-PV-07	SB-PV-09	SB-PV-10
Break	t = 0 s	t = 0 s	t = 0 s
Scram signal	P_{prim} = 12.97 MPa	P_{prim} = 12.97 MPa	P_{prim} = 12.97 MPa
Pump coastdown	Scram signal	Scram signal	Scram signal
Relief valve openings	P_{sec} = 7.82/8.03 MPa	P_{sec} = 7.82/8.03 MPa	P_{sec} = 7.82/8.03 MPa
High Pressure Injection System (HPIS)	CET = 623 K	-	-
Accumulator (ACC) discharge	P_{prim} = 4.51 MPa	P_{prim} = 4.51 MPa	P_{prim} = 4.51 MPa
SG depressurization	P_{prim} = 4 MPa	CET = 623 K	P_{prim} = 12.27 MPa + 30min
Low Pressure Injection System (LPIS)	-	P_{prim} = 1.24 MPa	P_{prim} = 1.24 MPa

In the three experiments, after the discharge of the accumulators, the valves are not isolated in order to study the influence of the intrusion of nitrogen into the primary system during the prolonged cooling of the core.

2.3. TRACE model

The model of the LSTF has been developed with the thermal hydraulic code TRACE5 patch 7 [16]. Fig. 1 shows the nodalization of the LSTF model using the Symbolic Nuclear Analysis Package (SNAP) software. The model is made up of 83 hydraulic components (1 VESSEL, 1 PRIZER, 2 PUMPS, 8 BREAKS, 11 FILLS, 15 VALVES, 22 TEES, 23 PIPES). The pressure vessel is modeled using the VESSEL 3D component. Both loops, the U-tube bundle, injection lines, accumulators, and steam generators are modeled using the PIPE component. To guarantee the boundary conditions present in the PIPES, the inlet and outlet temperatures, the pressure difference, and the heat transfer of the original tubes are preserved.

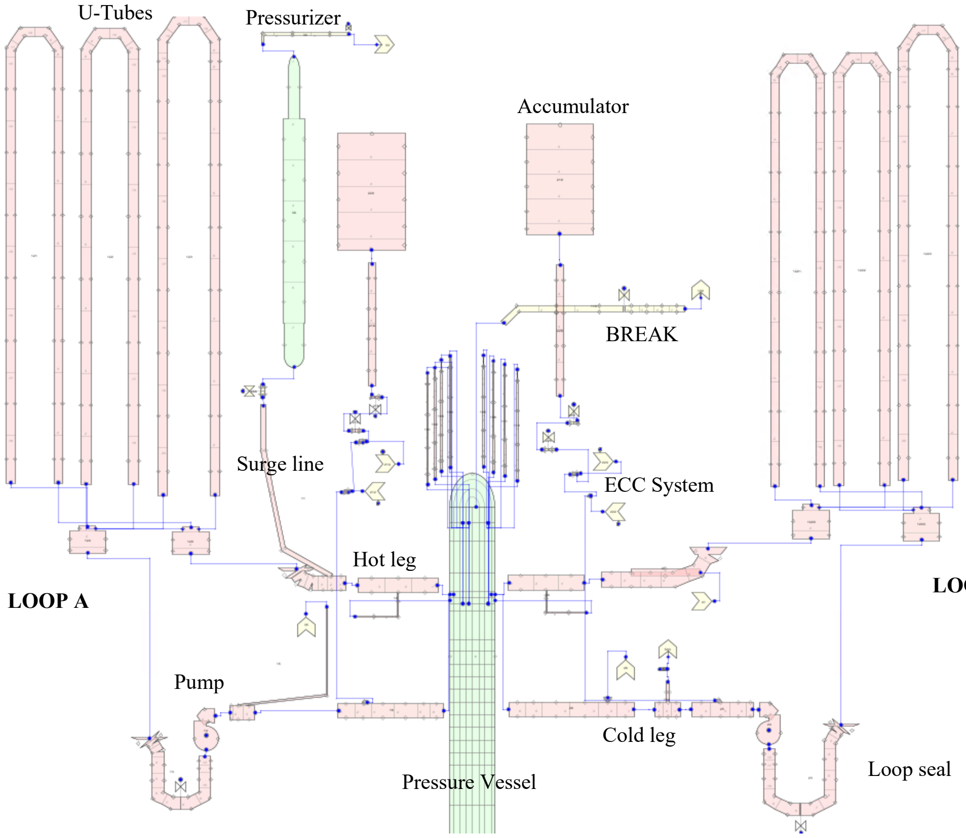

Fig. 1. Sketch of the LSTF model nodalization.

The power of the core is generated by the POWER component, which transfers this power to a series of electrical resistors modeled by Heat Structures (HTSTR) components. Each of these heat structures is associated with an axial level and radial sector of the core. In addition, the heat transfer that occurs between the primary and secondary circuit is also simulated with Heat Structures components. The main and auxiliary feed water systems are modeled using the FILL components, while the boundary conditions are simulated using VALVE and BREAK components. To measure the different variables, SIGNAL VARIABLE components are used, while TRIP and CONTROL BLOCK components are used to process these signals. This model has been previously verified in the simulation of different LOCA scenarios, with breaks of different sizes and locations.

3. RESULTS AND DISCUSION

The use of a verified model allowed the reproduction of the behavior of the main thermal hydraulic variables (pressures, temperatures, flow rates, liquid levels, etc.) and the chronology of the major events. However, the model needed to be updated to simulate the three experiments. The breaks have been modeled, and the control logic necessary for the operation of the safety systems involved in each test has been included. After reproducing the experiments SB-PV-07, SB-PV-09, and SB-PV-10, the same scenarios were then simulated, but isolating the accumulator tanks after discharge, preventing the entrance of nitrogen into the system.

3.1. Test SB-PV-07

In this test, although there is a temporary failure of the HPIS (High Pressure Injection System), its flow is available since the core is uncovered. Moreover, this coolant, together with that from the accumulators, is sufficient to recover the liquid level in the vessel and reduce the core temperature. The subsequent depressurization of both steam generators brings the facility to stable pressure and temperature conditions within 7000 s. Figure 2 shows a comparison between transients with (w N_2) and without (w/o N_2) nitrogen injection from ACC. As shown, the difference between the two simulations lies in the final conditions at which the primary system stabilizes, with pressure and temperature being slightly lower in the scenario in which nitrogen entry is prevented. Figure 2B plots the Maximum Peak Cladding Temperature (PCT) measured in the heat structures of the core. The increase of temperature indicates an uncover of the core.

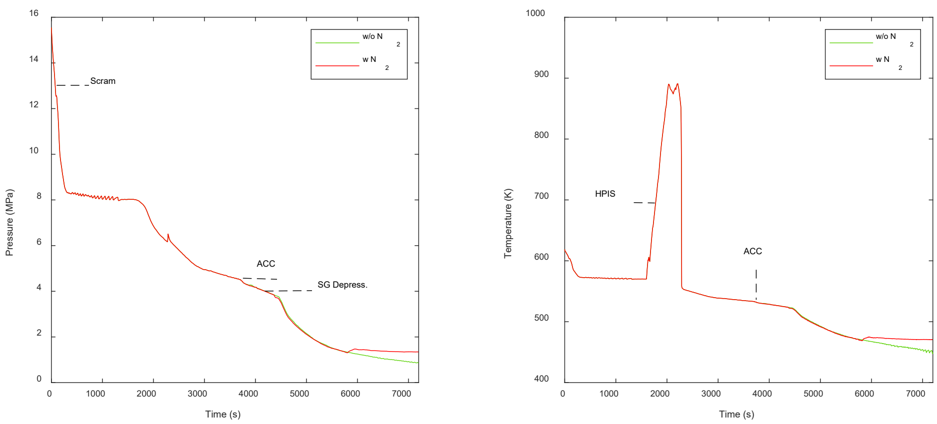

Fig. 2. SB-PV-07 scenario with and without nitrogen intrusion. A) Primary pressure. B) Maximum Peak Cladding Temperature (PCT).

Nitrogen from the accumulator tanks travels through the facility and, in the final phase of the experiment, accumulates in the cold legs and the steam generators. In the U-tubes, the nitrogen accumulation is higher in the downstream section (Fig. 3). This degrades heat transfer to the steam generators in the upper region of the U-tubes but does not prevent reflux cooling as the mass flow is not totally interrupted.

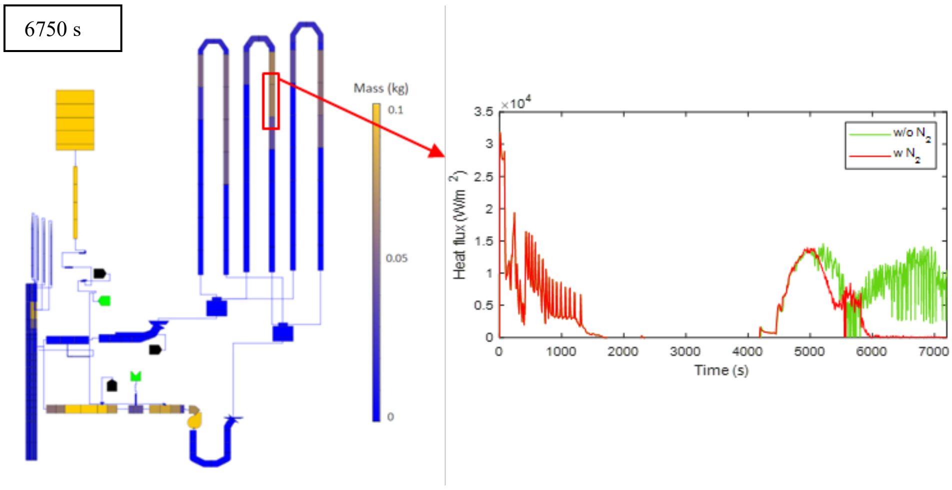

Fig. 3. Nitrogen distribution and heat flux at one U-tube steam generator node in SB-PV-07.

3.2. Test SB-PV-09

In the SB-PV-09 test, depressurization of the steam generators cannot prevent a very abrupt excursion of the PCT. In order to protect the heated rods, a temperature signal triggers a power reduction. A few seconds later, the discharge of coolant from the accumulator tanks recovers inventory in the facility and rapidly lowers the core temperature. Nitrogen intrusion impacts the activation of the LPIS (Low Pressure Injection System). First, nitrogen buildup in the steam generator tubes degrades heat transfer and hinders the depressurization of the system. This leads to a significant delay in the time of the LPIS injection, from 2470 to 3270 seconds, as well as a decrease in the flow rate supplied by the injection pumps (Fig. 4).

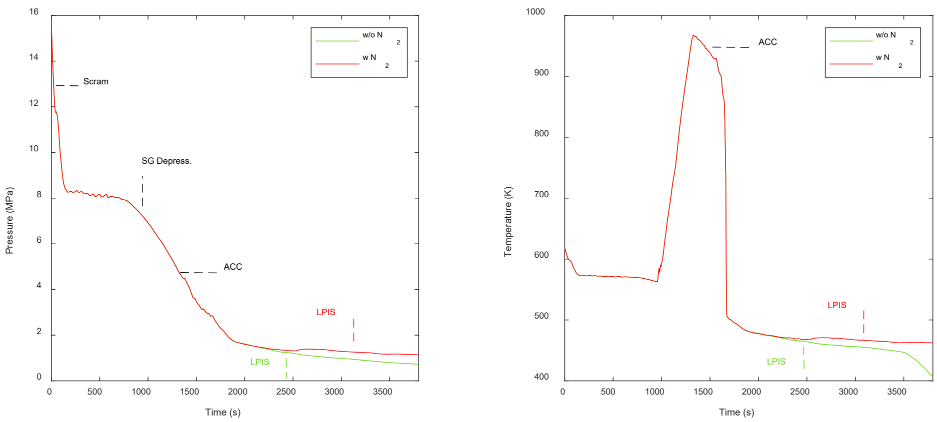

Fig. 4. SB-PV-09 scenario with and without nitrogen intrusion. A) Primary pressure. B) PCT.

Nitrogen buildup in the U-tubes has further effects. In this scenario, natural circulation ceases completely and reflux and condensation conditions cannot be achieved. Fig. 5 shows the accumulation of nitrogen in the U-tubes and the heat flux in a U-tube over the transient.

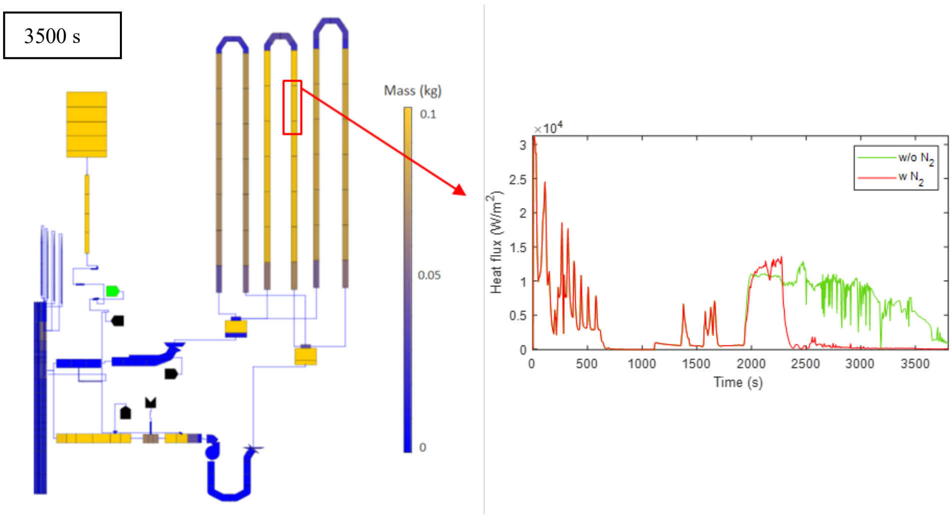

Fig. 5. Nitrogen distribution and heat flux at a steam generator node in SB-PV-09.

3.3. Test SB-PV-10

In this test, the first accident management measure to strengthen the decay heat extraction after the shutdown is an asymmetrical and controlled depressurization (55 K/h) at the steam generator in the loop without a pressurizer. Then, a natural circulation flow rate greater than 10 kg/s is established in that loop and avoids core heat up. Following coolant discharge from both accumulators at 4.51 MPa, nitrogen enters the primary system and modifies the natural circulation behaviours and depressurization (Fig. 6). As occurred in other experiments, the peak cladding temperature is also affected by nitrogen (Fig 7) and it does not decrease as in the absence of non-condensable gas.

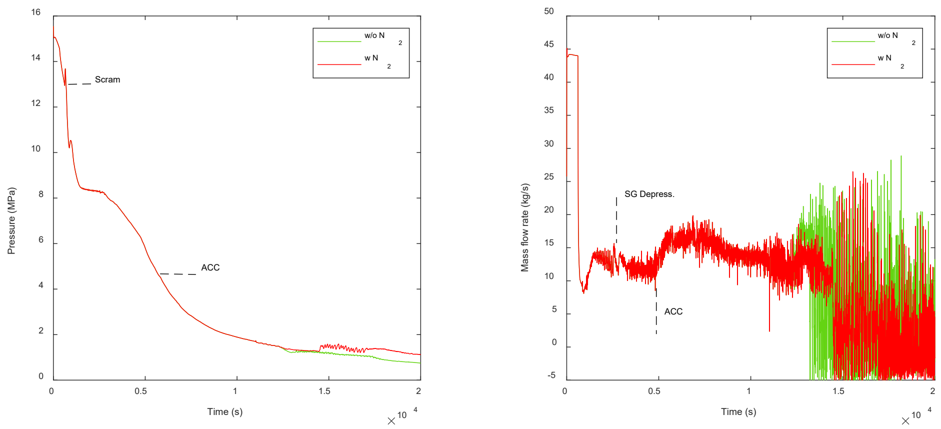

Fig. 6. SB-PV-10 scenario with and without nitrogen intrusion. A) Primary pressure. B) Mass flow rate in loop without pressurizer.

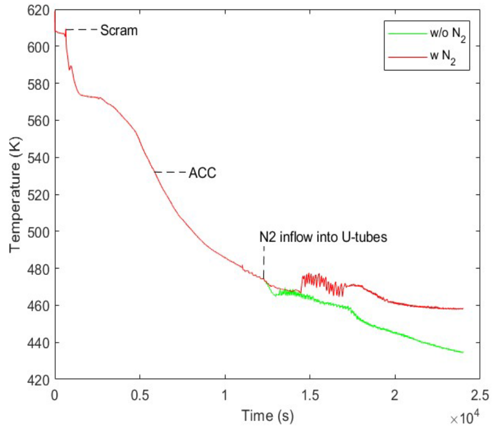

Fig. 7. SB-PV-10 scenario with and without nitrogen intrusion. Peak Cladding Temperature.

Nitrogen is distributed according to the thermal hydraulic conditions in the facility. As shown in Fig. 8, in which the pressurizer is located, the nitrogen remains stagnant in the cold leg. In the non-pressurized loop, the non-condensable gas travels to the U-tubes and accumulates.

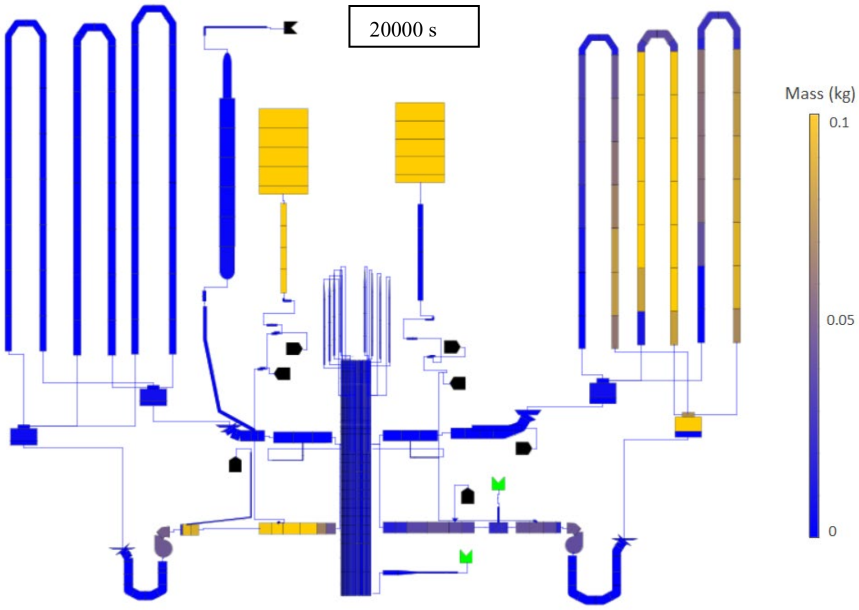

Fig. 8. Nitrogen distribution in the SB-PV-10 test.

4. CONCLUSIONS

Nitrogen gas intrusion during SBLOCA scenarios at the reactor coolant system impacts the thermal-hydraulic performance, and the experiments at the LSTF facility reveal some effects. This work analyzes and simulates with TRACE5 three tests performed in LSTF, aimed at studying the consequences of nitrogen intrusion due to the isolation failure of the accumulator valves. These scenarios are compared with equivalent ones in which the tanks remain isolated and nitrogen inflow is avoided. Results confirm that nitrogen buildup in the U-tubes severely degrades heat transfer efficiency, which can result in the cessation of natural circulation and prevent reflux cooling. Overall, these findings underscore the importance of developing robust accident management strategies to mitigate the adverse effects of nitrogen gas intrusion. Preventing nitrogen from entering the primary system after the discharge of the accumulators enhances the effectiveness of core cooling mechanisms and the stability of the reactor system under accident conditions.

ACKNOWLEDGMENTS

This work has been developed under the project “*Análisis del impacto de la intrusión de gas nitrógeno en el circuito primario de refrigeración de plantas nucleares*” (SUBV-6/2023, *Subvenciones destinadas a la realización de proyectos de I+D+i relacionados con las funciones del CSN*). The authors are grateful to the Spanish Safety Council (CSN) that has financed this project.

REFERENCES

[1] P. Kral. Sources and effect of non-condensable gases in reactor coolant system of LWR, NURETH-16, (2015).

[2] B. Noel, R. Derauz, Reflux condenser mode with non-condensabel Gas: assessment of CATHARE against BETHSY Test 7.2C. *Nuclear Engineering and Design* 149, (1994).

[3] K. Umminger, L. Dennhardt, S. Schollenberger and B. Schoen, Integral test facility PKL: Experimental PWR accident investigation, *Science and Technology of Nuclear Installations*, 16, (2012).

[4] T. Takeda and I, Ohtsu, ROSA/LSTF test on nitrogen gas behavior during reflux condensation in PWR and RELAP5 code analyses, *Mechanical Engineering Journal*, The Japan Society of Mechanical Engineers (2018).

[5] T. Takeda, A. Ohnuki, D. Kanamori and I. Ohtsu, ROSA/LSTF Tests and RELAP5 Post test Analyses for PWR Safety System Using Steam Generator Secondary-Side Depressurization against Effects of Release of Nitrogen Gas Dissolved in Accumulator Water, *Science and Technology of Nuclear Installations*, (2016).

[6] T. Takeda and I. Ohtsu, ROSA/LSTF Tests and Posttest Analyses by RELAP5 Code for Accident Management Measures during PWR Station Blackout Transient with Loss of Primary Coolant and Gas Inflow, *Science and Technology of Nuclear Installations* (2018).

[7] T.-J. Liu, Reflux condensation behavior in a U-tube steam generator with or without noncondensables, *Nuclear Engineering and Design,* 204 (1–3), 221–232, (2001).

[8] K.-W. Lee, H.C. No, I.-C Chu, Y.M. Moon, M.-H. Chun, Local heat transfer during reflux condensation mode in a U-tube with and without noncondensible gases, *International Journal of Heat and Mass Transfer,* 49 (11–12), pp. 1813–1819, (2006).

[9] S. T. Revankar and D. Pollock, Laminar film condensation in a vertical tube in the presence of noncondensable gas, *Applied Mathematical Modelling*, Volume 29, Issue 4, pp. 341–359 (2005).

[10] I. K. Huhtiniemi and M. L. Corradini, Condensation in Presence of Non-condensable Gases. *Condensation and Condenser Design*, ASME (1993).

[11] G. Caruso, F. Giannetti and A. Naviglio, Experimental investigation on pure steam and steam–air mixture condensation inside tubes, *International Journal of Heat and Technology* 30, pp. 77-84 (2012).

[12] T. Takeda, 2018. Data Report of ROSA/LSTF Experiment SB-PV-07 - 1% Pressure Vessel Top Break LOCA with Accident Management Actions and Gas Inflow. JAEA-Data/Code 2018-003.

[13] T. Takeda, 2018. Data Report of ROSA/LSTF Experiment SB-PV-09 – 1.9% Pressure Vessel Top Break LOCA with SG Depressurization and Gas Inflow. JAEA-Data/Code 2021-006.

[14] T. Takeda, 2007. Data Report of ROSA/LSTF Experiment SB-PV-10 – 0.1% Pressure Vessel Bottom Small Break LOCA Experiment.

[15] The ROSA-V Group. ROSA-V Large Scale test facility (LSTF) system description for the third and fourth simulated fuel assemblies. Tech. Rep. JAERI-Tech 2003-037, Japan Atomic Energy Agency (2003).

[16] United States Nuclear Regulatory Commission, TRACE V5.0: Theory Manual Field Equations, Solution Methods, and Physical Models (V50rc3) (2007).