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Analyses of the MELCOR capability to simulate integral PWR using passive systems in a DBA scenario☆

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ABSTRACT

This paper is focused on a thermal–hydraulic transient analysis of a generic 300 MWe integral Pressurized Water Reactor (iPWR) based on a passive safety mitigation strategy and dry containment. The main target of the paper is to analyze the MELCOR code's capability to simulate the operation of the passive systems in Small Modular Reactor (SMR) configuration and the consequent plant behavior. The nodalization and the modeling approach used in the MELCOR code have been described in the present work, and a double-ended rupture of the Direct Vessel Injection (DVI) line has been postulated as initiating event.

The results of the analysis demonstrated that the MELCOR code can qualitatively replicate the dominant phenomena that drive the passive mitigation strategy's operation. Additionally, the core reflooding was made possible by the operation of the available safety systems, which was adequate to prevent severe accident conditions during the entire simulated transient. The activity has been developed in the framework of SASPAM-SA Horizon Euratom project as the base for further derivative activities focused on assessing the capability of the MELCOR code to simulate the phenomena taking place in postulated plausible severe accident scenarios in iPWR.

1. Introduction

The implication of climate change, paired with the increase in energy demand and the most recent interest in energy security and independence, results in a renewed openness toward nuclear energy and new advanced reactor designs. Small Modular Reactors (SMR) are one of the most promising options for the deployment of new generation reactors, which aim to satisfy the main goals of high grade of safety, inherently increasing the safety of the plant, and the need to improve economic efficiency, reducing capital costs and construction times (Maccari et al., 2021).

SMRs are particularly interesting in this framework as the integral Pressurized Water Reactors (iPWR) are seen as complementary and synergic to large Light Water Reactor (LWR). Regarding safety, this new integral design results in a decreased core damage frequency thanks to the smaller size of the SMRs, which gives some significant opportunity: less nominal power means less residual power to be removed in an accidental scenario, making the reactor design simpler and enhancing the possibility of using passive safety systems instead of active systems (Maccari et al., 2021). iPWRs strongly benefit from the operational

Acronym: ADS, Automatic Depressurization System; BDBA, Beyond Design Basis Accident; DBA, Design Basis Accident; DiD, Defence-in-Depth; DVI, Direct Vessel Injection; EBT, Emergency Borated Tank; EHRS, Emergency Heat Removal System; EPZ, Emergency Planning Zone; HCP, High Containment Pressure; HX, Heat eXchanger; IAEA, International Atomic Energy Agency; IPWR, integral Pressurized Water Reactor; LOCA, Loss of Coolant Accident; LDPC, Low Differential Pressure RC-Containment; LPL, Low PRZ Level; LWR, Light Water Reactor; LGMS, Long-term Gravity Make-up System; MASLWR, Multi Application Small Light Water Reactor; NPP, Nuclear Power Plants; PRZ, Pressurizer; PSS, Pressure Suppression System; QT, Quench Tank; RI-DC, Riser-Downcomer; RC, Reactor Cavity; RCS, Reactor Coolant System; RPV, Reactor Pressure Vessel; RWST, Refueling Water Storage Tanks; SA, Severe Accident; SASPAM-SA, Safety Analysis of SMR with PAssive Mitigation strategies – Severe Accident; RCP, Reactor Coolant Pump; SG, Steam Generator; SMR, Small Modular Reactor; SNL, Sandia National Laboratory; USNRC, US Nuclear Regulatory Commission.

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experience of large LWRs and incorporate evolutionary features that enhance intrinsic safety, such as the integration of passive safety systems (IAEA, 2022). Many iPWR designs have been developed in relatively recent years such as SMART (Park, 2021), IP200 (Jiang, 2019), ACP100 (Zhu, 2016), NuScale (Esmaili, 2018), IRIS (IRIS International Reactor Innovative and Secure, IRIS Plant Overview, 2002) and NUWARD (Francis, 2024). Developed by KAERI, SMART is an iPWR with a thermal power of 365 MWth. In addition to producing energy, this reactor was built to provide heat for the desalination process. The Harbin Engineering University created IP200, another iPWR. Its 12 integrated once-through steam generators remove power from the main system, and its thermal power is approximately 220 MWth. It exploits passive systems as protection in accidental sequences. NuScale is also an iPWR, conceived by NuScale Power Inc., and is the first to be licensed by the US NRC in 2020. IRIS is an iPWR concept ideated by an international team lead by Westinghouse. Its power output is 330 MW_e and it also relies on passive safety systems for accidental sequences protection. Other light water SMR designs as the AP300 (non-integral PWR) (AP300 SMR) and BWRX-300 (BWR) (Matsuura, 2023) are promising designs in the SMR field.

When performing safety analyses for current and advanced reactors, the characterization of thermal–hydraulic phenomena is crucial, which, combined with the best practices already in use to model traditional large-scale reactors (PWR and BWR) allows an accurate reproduction of the involved phenomena. However, some characteristics of advanced SMRs allow for differentiation between the two in terms of phenomenology and development of plausible accidental scenarios. These features (Mascari et al., 2023) are related mainly to the:

- Containment processes and interaction with the Reactor Cooling System (RCS), e.g. strong coupling effect and feedback between the primary and the containment;
- Low pressure phenomena;, natural circulation phenomena (e.g. interaction among parallel circulation loops inside and outside the vessel and influence of non-condensable gases), steam liquid interaction phenomena (e.g. direct condensation), gravity driven reflood phenomena (e.g. heat transfer coefficient) and liquid temperature stratification phenomena (e.g. vessel lower plenum and downcomer).
- Phenomena related to new system components or reactor configuration, e.g. integral primary system, helical coil steam generators, natural-circulation heat removal systems, etc.

The present work is preparatory for the Horizon Euratom Safety Analysis of SMR with PAssive Mitigation strategies – Severe Accident (SASPAM-SA) project (European Commission,2023), coordinated by ENEA (Italy) (SASPAM-SA); (Horizon Euratom Safety Analysis of SMR); (Mascari, 2022). The project aims at investigating the applicability and transfer of the operating large-LWR reactor knowledge and know-how to the near-term deployment of iPWR, in the view of Severe Accident (SA) and Emergency Planning Zone (EPZ) European licensing analyses needs.[1] This research is focused on characterizing the MELCOR 2.2 computer code capability in predicting the main thermal–hydraulic phenomena occurring in an iPWR using a passive mitigation strategy in a Design Basis Accident (DBA) scenario. MELCOR is developed by Sandia National Laboratories (SNL) for the USNRC (Humphries et al., 2024). Recently, developers added several features to facilitate the simulation of SMR-specific phenomena (Humphries et al., 2024a,b), such as the implementation of heat transfer correlations for helical-coil SGs.

Studies in high power reactors highlight the MELCOR capability to simulate DBAs in passively protected reactors or facilities, such as the AP1000 (Malicki, 2019); (Seyffert and Malicki, 2024); (Włostowsky and Darnowski, 2024). Later in the paper is noticeable how this reactor shares many similarities with the generic chosen iPWR, highlighting that previous LWRs analyses and experiences can play a major role in modern reactors' code validation and best practices. Also, ongoing research is considering the applicability of a variety of codes for SMRs (Maccari et al., 2021); (Mascari et al., 2023); (Malicki and Lind, 2024); (Fakhraei, 2020) due to their modeling peculiarities. Indeed, the above mentioned SMRs features such as helical coil steam generators, are critical to assess, causing interest to conduct more specific studies.

In the first part of this work, a model with MELCOR code has been developed, followed by steady-state and transient analysis under DBA scenarios. This work will be used as a starting point in the Horizon Euratom SASPAM-SA project to analyze postulated plausible Beyond Design Basis Accident (BDBA) and SA sequences with MELCOR code.

Currently, in Europe there is a growing interest in the deployment of SMRs, and several activities are underway in many countries preparing for possible licensing needs. The SASPAM-SA project's key outcomes are to support the iPWR licensing process by bringing up key elements of the safety demonstration needed and to speed up the licensing and siting process of iPWRs in Europe. The SASPAM-SA project proposal has been submitted and has been founded in HORIZON-EURATOM-2021-NRT-01–01, "Safety of operating nuclear power plants and research reactors", it is coordinated by ENEA (Italy) and twenty-three Organizations from fourteen Countries are involved with a duration of 4 years.

To maximize the knowledge transferability and impacts of the project, two generic design concepts, have been selected for the analyses characterized by different evolutionary innovations in comparison with the larger operating reactors:

- Design 1: iPWR characterized by a submerged containment and electric power of about 60 MWe.
- Design 2: iPWR characterized by the use of several passive systems, a dry containment and an electric power of about 300 MWe.

These two generic reactor concepts include the main iPWR design features, considered the most promising designs ready to be deployed on the European market, allowing to assess in a wider way the capability of codes (SA and CFD) to simulate the phenomena typical of iPWR. It is not

[1] Advanced designs, as iPWRs, are in general characterized by common features with the current operating large-LWR and by other specific features typical of their inherent evolutionary designs, providing safety advantages that reinforce the first three levels of the DiD principle. However, even if a plant is designed with advanced inherent features (through the reinforcement of the 1,2,3 DiD levels) allowing a reduction of the Core Damage Frequency (CDF) (Carelli et al., 2004), independent features for preventing and mitigating a severe accident sequence have to be included in its design (DiD level 4) together with the offsite emergency response (DiD level 5). Therefore, some scenarios that could lead to severe accidents need to be postulated and deterministically studied. More detailed information's in Horizon Euratom Safety Analysis of SMR; SMR Regulators' Forum Pilot Project Report (2018). Therefore, it is necessary to assess the capability of internationally recognized SA code to describe the behaviour of the most promising iPWR designs during SA scenarios and to predict the resulting radiological impact on– and off-site.

the project's objective to assess the generic reactor designs selected but based on the project findings, it will allow a more general statement on the code's applicability to designs that possess common features under postulated SA conditions. No PSA considerations will be made in the project due to the generic nature of the reactor concept considered, then the scenarios identified will be characterized in terms of severity but not in terms of probability (Mascari, 2022).

2. Generic IPWR design considered and model considerations

In order to assess the accuracy of a code, an assessment database (Mascari et al., 2015) is generally needed. Some studies can be highlighted regarding the assessment of system and integral SA codes to simulate the code's capability against passive system phenomenology. For example, the extensive assessment of the TRACE code against the MASLWR database, particularly to simulate natural circulation in integral configurations, the heat transfer with helical coil steam generators, and primary/containment coupling is reported in (Mascari et al., 2016) together with assessing the scaling issue for natural circulation (Mascari et al., 2024). In relation to these phenomena, the IAEA International Collaborative Standard Problem (ICSP) on integral PWR design, can also be mentioned (International Atomic Energy Agency, 2013), where different state-of-art codes have been tested against IAEA-ICSP MASLWR data. Regarding the analyses of the capability of state-of-the-art tools, it is important to highlight the PERSEO benchmark (Mascari, 2023), developed in the WGAMA framework, where different state-of-art codes, including the MELCOR code, have been used to simulate a full-scale passive heat removal system.

The target of this activity is not to conduct a quantitative code assessment. Instead, we qualitatively analyze the code's ability to predict the expected phenomena, using previous activities related to the design of the SPES-3 facility to identify the phenomena that drive the phenomenology. Clearly, further studies with experimental data will be necessary to achieve a comprehensive assessment of the code's capabilities from both a qualitative and quantitative perspective.

The activity presented here has been developed in the frame of the WP2, Input deck development and hypothetical SA scenarios assessment, coordinated by KIT (Germany). The major objective of this WP will be the development of generic but representative iPWR SA code input-decks and the analysis of the iPWR behavior under hypothetical postulated SA conditions. The studied plant is a generic iPWR characterized by the use of several passive systems, a dry containment, and an electric power of about 300 MWe, which in the framework of the SASPAM-SA project is named Design 2 (SASPAM-SA).

The Design 2 configuration permits the elimination of the occurrence of some accident sequences. The integral RPV contains, other than the core, all the major RCS components, which include: reactor coolant pumps (RCPs), compact Steam Generator (SG), e.g. helical coil bundles once-through SG, the pressurizer (PRZ) and heaters. Therefore, large Loss Of Coolant Accident (LOCAs) are eliminated by the integral layout of the RPV and only Small break LOCA will be studied.

Passive systems are, on the one hand, an advanced solution to increase the inherent safety of the plant (e.g., there is no need for pumps), on the other hand:

- They require extensive investigation for assessing the functional failure related to the thermal–hydraulic phenomena driving the operation of the systems and assess the related uncertainties,
- May still need an active initiation, and
- Are characterized by very limited operational experience (to have more detailed information (European SMR).

Considering this, the impact of the operation of passive systems on postulated SA progression is a novel topic and needs to be assessed (Maccari et al., 2021). Indeed, the specific assessment of the code's capability to simulate passive systems phenomenology is crucial. Undoubtedly, in LW-SMR reactors the evolution of BDBA/SA scenarios depend on the postulated partial or total unavailability of passive systems (e.g. postulate the failure of passive system valve activation). Therefore, it is necessary to analyze the capability of SA codes to predict the main thermal–hydraulic phenomena involved in the passive mitigation strategy of LW-SMR to adequately apply them for the design of accident management's strategy (the first phase of a SA is only dominated by thermal–hydraulic phenomena, then in-vessel is affected by thermal–hydraulic phenomena).

A reference database was needed to develop the MELCOR input-deck. Considering the Design 2 reactor type characteristics and the selected passive systems (Passive decay heat removal, Primary automatic depressurization, Passive high-pressure safety injection, Containment pressure suppression, Passive low-pressure safety injection and Flooding of the Reactor Cavity), a generic IRIS SMR type has been here considered as a reference for this analysis. During the MELCOR nodalization of the generic IRIS design, no proprietary data have been used; therefore, the main geometric information has been determined by scaling the data available from the SPES-3 facility (Maccari et al., 2021; Ferri and Congiu, 2008), by engineering evaluation or public general data available for the IRIS reactor (Carelli et al., 2004). The plant specifications and design details have been collected in SASPAM-SA (2023).

Fig. 1 shows the integral RPV and the primary coolant flow path. First, water flows through the core and enters the riser region (defined by the extended core barrel). The coolant is then directed into the upper annular plenum, where the suction of the RCPs is located, and the flow is directed downward through its associated helical coil SG module. The flow path continues downstream through the annular downcomer region to the lower plenum and then back to the core, completing the primary coolant flow path. The RPV is placed inside a reactor containment spherical shell, designed to sustain high pressure as part of the reactor's passive safety approach and mitigation strategy. Every passive safety system is located inside the spherical containment, except for the Passive Decay Heat Removal system.

Since the activity aims to test the MELCOR code capability in reproducing the different passive safety systems behavior and their interaction in a SMR configuration, the following passive systems have been considered (Fig. 1, the scheme shows only one line for each redundant system) to have a wider code capability analysis:

- Passive decay heat removal (2x100% redundant: two tanks, four Heat Exchangers (4x50%) in total): this system is designed to remove the core power from the primary coolant to the environment during accidental conditions. The system includes a Heat eXchanger (HX), submerged in an elevated pool, connected to the secondary side of the SGs. In a generic IRIS design, this system corresponds to the Emergency Heat Removal System (EHRS), and the elevated pool located outside the containment is the Refueling Water Storage Tanks (RWST).
- Primary automatic depressurization: This system ensures the RPV fast depressurization in an automatic way to enable the passive safety injection of make-up water. This system in a generic IRIS design corresponds to the Automatic Depressurization System (ADS), which is composed of two stages. The opening of the ADS stage 1 valves leads to the dumping of steam on the Quench Tank (QT) located in the containment's DryWell (DW). While the ADS stage 2 discharges the steam directly inside the DW.
- Passive high-pressure safety injection (2x100% redundant, two tanks): this system provides an high-pressure source of borated water in case of LOCA. It is composed of a tank inside the containment and connected at the top and bottom with the RPV. In a generic IRIS design this system corresponds to the Emergency Borated Tank (EBT), which is connected to the RPV, allowing natural circulation to occur between the RPV and the tank once activated.

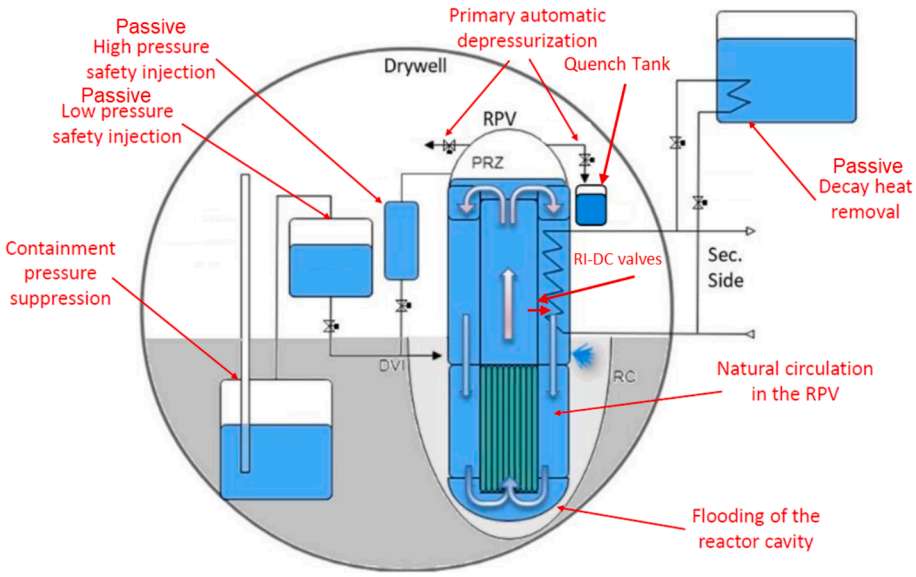


Fig. 1. Conceptual scheme of the generic iPWR considered.

- Containment pressure suppression (2x100% redundant, two tanks): this system is designed to provide a means of condensing steam released into the containment. The system is composed of large tanks located inside the DW volume and connected to the DW atmosphere through vent pipes. In a generic IRIS design, this system corresponds to the Pressure Suppression System (PSS) tank.
- Passive low pressure safety injection (2x100% redundant, two tanks): this system can establish and maintain the core in a safe shutdown condition following a LOCA by injecting long-term gravity makeup water. It is composed of elevated tanks, located in the DWand connected to the RPV via the DVI lines. In a generic IRIS design it corresponds to the Long-term Gravity Make-up System (LGMS).
- Natural circulation in the integral RPV: under accidental conditions, following the primary pumps coast-down, natural circulation inside the integral RPV occurs. To allow natural circulation at a lower RPV water level in case of LOCA, additional connections between the riser and the downcomer SGs can be considered to allow the primary water to flow through a shorter path, bypassing the pumps suction plenum. In the generic IRIS design assumed, the reactor is provided with Riser-Downcomer (RI-DC) valves kept closed by the primary pumps' delivery pressure in nominal reactor conditions and can automatically open once the pumps are turned off.
- Flooding of the RC: the flood-up of the RC ensures that the Lower Head (LH) is surrounded by water following any LOCA event. This may be fundamental for in-vessel melt retention in case of Severe Accident (SA), and once the water level reaches the DVI elevation, it can provide long-term gravity makeup so that the RPV water inventory is maintained above the core.

3. Description of the MELCOR nodalization

The input deck for the generic iPWR considered was developed with MELCOR 2.2.18019. The input deck is composed of an executive driver and several major packages that together model the major systems of a reactor plant and their generally coupled interactions (Nuclear Regulatory Commission, 2014). These packages form the main structure of the reactor nodalization and simulate the thermal–hydraulic behavior of the generic iPWR considered during normal operation and accident conditions.

A detailed MELCOR thermal–hydraulic nodalization of the reactor has been developed and contains 121 Control Volumes (CVH), 153 Flow Paths (FL), and 122 Heat Structures (HS). The model is made of 4 subsystems: the primary circuit, the secondary circuit, the passive safety systems and the containment, as shown in Fig. 2. Those systems are coupled to each other both thermally, with heat structures, and hydrodynamically with flow junctions to reproduce the response of the reactor to various accidental scenarios. More details on primary/containment coupling and related experiments, publicly available to the international community, are in (Ferri and Congiu, 2008); (Carelli et al., 2004); (Achilli et al., 2012).

3.1. Thermal-hydraulic nodalization of RPV (primary circuit) and secondary circuit

For the RPV nodalization all the integrated circuit has been implemented. The eight SGs modules on the primary side are modelled as an equivalent one. The downcomer SGs are connected to the riser region through Riser-DownComer check (RI-DC) valves, designed to facilitate the natural circulation under low RPV water level condition, allowing the primary water to flow through a shorter path bypassing the pump's suction plenum.

The same approach used for the primary side SGs was used to model the fluid in the helical coil SGs, with 9 vertical volumes coupled with the primary SGs volumes. To reproduce the thermal–hydraulic characteristics of the helical coil, the SGs tubes are modeled with cylindrical inclined heat structures. On the tube side, the dedicated helical coil model for heat transfer is used, while for the shell side the Zukauskas model is selected (Beeny et al., 2020); (Zukasukas, 1972).

3.2. Nodalization of the core and lower head

The core and lower head of the reactor have been modeled using the 2D cylindrical axial-symmetric geometry with the COR package, coupling the COR mesh with the respective hydrodynamic volumes. The core mesh has been chosen with a higher grade of detail compared to the hydrodynamic volumes to have an accurate model of the core behavior under accident scenarios. In particular, the COR mesh is divided into 16 axial meshes and 6 radial concentric rings as shown in Fig. 3. The active core is between axial levels 8–14 and rings 1–4. The lower plenum instead has been nodalized with 5 axial regions and radially by 6 concentric rings.

During the modelling of the LH with MELCOR 2.2 was necessary to modify the original vessel geometry. Originally the junction between the

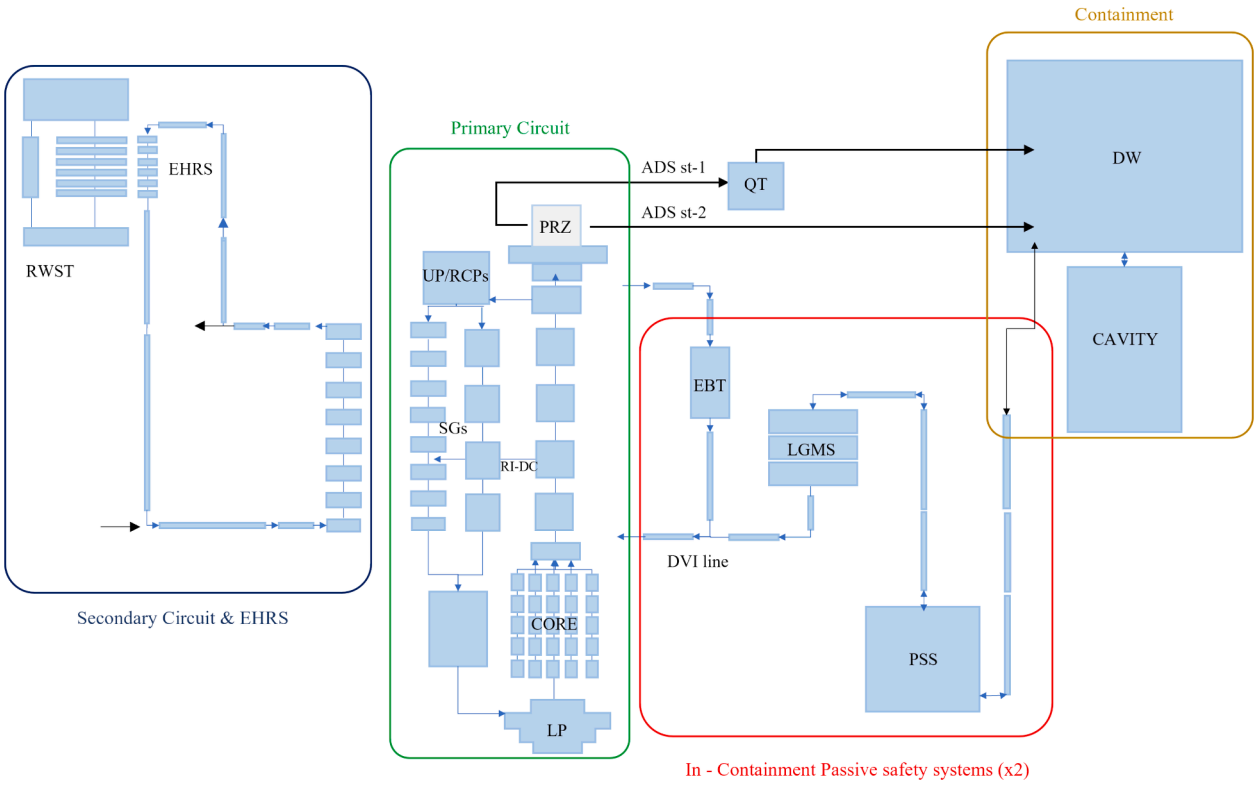


Fig. 2. Thermal-hydraulic nodalization adopted in the MELCOR model of the generic iPWR Design 2.

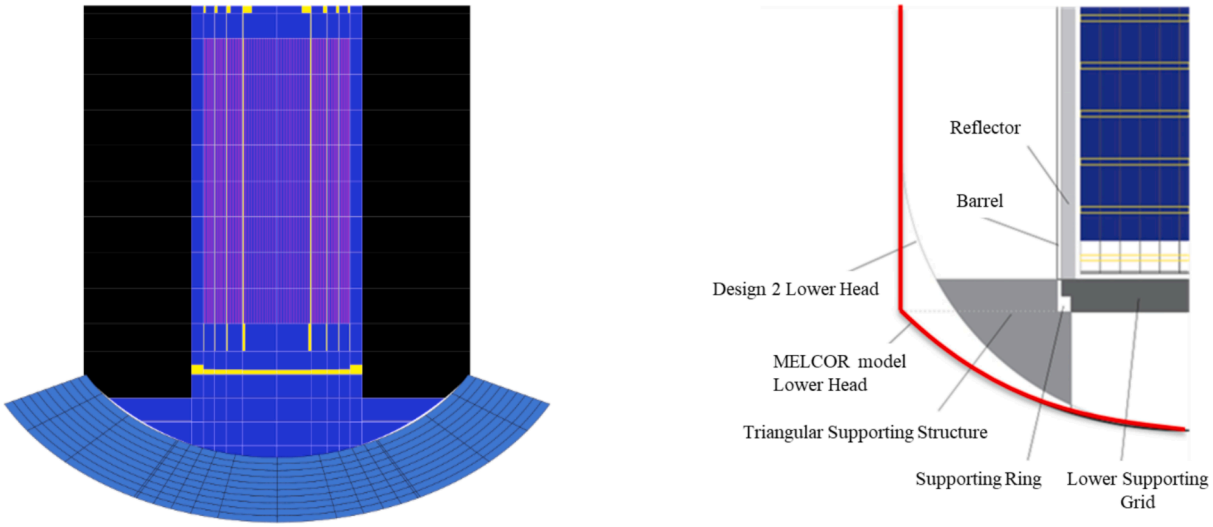


Fig. 3. Design 2 MELCOR COR (left) and lower head (right) models.

hemispherical section and the cylindrical part of the vessel was placed above the elevation of the lower supporting plate. However, MELCOR 2.2 allows the modeling of the lower head at a maximum elevation below the bottom of the lower plate. This force changes the original LH geometry, lowering the junction of the hemispherical head under the lower plate and increasing the total volume of the lower plenum as shown in Fig. 3.

3.3. Thermal-hydraulic nodalization of the passive safety systems

The in-containment safety systems lines, which include the LGMS, EBT, and PSS, have been modeled separately, despite being identical, since a different behavior is expected between them during the considered transient sequence (one line presents the broken DVI in case of DVI rupture). The EHRS-RWST exchanger modules are located out of the containment, and they are collapsed in one equivalent HX module divided into six vertical volumes. They are also thermally coupled with six vertical volumes belonging to the RWST. Boundary conditions are applied to the RWST to consider the heat and mass losses to the environment and keep its pressure at atmospheric value.

3.4. Containment thermal-hydraulic

The RPV is placed in a spherical containment filled with inert gas

(Nitrogen) as seen in Fig. 1. The volume is divided into 2 main zones:

- The DW, in the upper part, consists of a single volume;
- The Reactor Cavity (RC), a concrete lower containment volume designed to be flooded for mitigation purposes.

Each PSS tank is connected to the DW atmosphere through a vent pipe (Fig. 2) performing the containment pressure control. The upper volume of the DW is connected to the RPV via the ADS-2 line, with a normally closed valve. Instead, the lower volume is connected to the QT via the ADS-1, which downstream of a closed valve discharges into the pressure suppression system. The RC is located under the DW and is made by a single CVH that collects the liquid break flow and any condensate from the containment.

4. Calculation results and discussion

4.1. Steady state calculation results

As a starting point for transient analysis, the nominal steady state conditions have been simulated. To match the nominal conditions, the primary pump's head is set up at nominal value and the PRZ pressure regulation is simulated to speed up the stationary phase, which lasts 2000 s. In the present preparatory work, the reference steady state values are taken from Maccari et al. (2021).

4.2. DBA transient simulation

The Small Break LOCA (SB-LOCA) analyzed, concerning the double-ended break of one DVI lines, is the lowest possible elevation for a LOCA and, hence the most challenging accidental scenario in terms of safety. Since the injection systems are fully redundant, two DVI lines are modelled. We will refer to them as DVI line-A and DVI line-B and their respective safety systems with corresponding letters (ex: LGMS-B is the low injection system injecting in DVI line-B). Considering the DVI line-A as the broken line, the accident consists of 2-inch equivalent double-ended guillotine break of the DVI line. In this DBA transient progression, the availability of all the emergency passive safety systems is assumed. According to (Maccari et al., 2021), the reference DBA sequence of the iPWR considered can be divided into 3 Phenomenological Windows (PhW).

PhW1, the blow-down phase, is the period right after the break during which the primary systems pressure is reduced and the containment pressure increases until the reactor coolant system and containment pressure are equalized for the first time. The pressure increase inside the containment is limited by the PSS, while the depressurization of the RPV takes place thanks to the opening of the ADS-stage 1 valves and the decay heat removed by the EHRS. Once the pressure in the RPV and the containment are equalized for the first time the blow-down phase is considered concluded, at this point the flow from the break is reduced and the LGMS-B tank starts to inject water via the DVI line-B.

PhW1 is followed by PhW 2, which is called the depressurization phase. During PhW2, the primary circuit pressure decreases, and many phenomena take place in the DW due to its interaction with the PSS tanks. After the opening of the ADS-stage 2 the pressures of the containment and the RPV are coupled and can start the PhW 3, the cooling phase, where pressure and core decay heat slowly decrease.

4.3. DBA calculation results

The MELCOR simulation of the DBA has a total simulation time of 72 h, to reproduce the behavior of the passive safety systems during the transient simulation. The computational time for running the simulation resulted in around 96 h. The result of the MELCOR simulation is described and discussed here, focused on the principal events characterizing the passive mitigation strategy and describing the main thermal–hydraulic phenomena qualitatively predicted by the code along the transient. The sequence of main events characterizing the transient simulation is summarized in Table 2.

PhW 1 starts with the break opening at 0 s and lasts for the first 2200 s (after the LDPC signal is triggered), causing the RPV depressurization and consequently the containment pressurization. The secondary circuit and the reactor core are still operating at nominal reference power until the containment pressure reaches the setpoint for the High Containment Pressure (HCP) and the SCRAM signal is triggered, at 27 s. At this point the core fission power is shutdown. The flow through the break causes a fast depressurization of the RPV, with a peak value of about 160 kg/s (Fig. 4). In Fig. 5 the calculated trend of the PRZ, DW, PSS and secondary circuit pressure is reported. During the initial phase (from 0 to 1500 s) a progressive increase in the containment pressure is observed, and both steam and non-condensable gases from the DW volumes are dumped into the PSS tanks. The condensation in presence of non-condensable gases is modelled in the MELCOR HS package, which impacts condensation on the DW walls negatively (Yoon et al., 2017); (Humphries et al., 2024). The PSS tanks, which are connected to the LGMS tanks via the PSS vent lines, follow the containment pressurization trend.

The SCRAM signal triggers the secondary circuit isolation and the openings of the EHRS valves at 27 s. Due to the isolation of the secondary circuit and the delay time for the EHRS valve opening an increase in the SGs pressure is observed reaching a peak value of 10.5 MPa. Once the EHRS valves open, a fast depressurization takes place thanks to the natural circulation between the EHRS-HX and the SGs. The decay power from the core is transferred into the RWST, condensing the steam coming from the EHRS hot leg, which increases the pool temperature gradually during the transient. The Low PRZ Level (LPL) signal is triggered at 100 s, starting the primary pumps to coast down and the RI-DC valves to open, in order to facilitate the natural circulation inside the RPV. During the initial phase of the transient, till the water level covers the pump suction, two different natural circulation paths are established. The first path flows along the riser until the pump suction and descends through the downcomer around the SGs tube bundle. The second path flows at middle-riser height through the RI-DC check valves, descending along half of the SG, cooling the primary water. The total power removed from the primary circuit through the SGs is shown in Fig. 6 together with the core decay heat and the power transferred from the EHRS to the RWST. Some oscillations in the SGs removed power are detected, which are due to the oscillating water mass flow through the RI-DC valves path. This is addressed later in the paper.

Table 1
Steady state calculation values.

Parameters	Reference Value (Maccari et al., 2021)	MELCOR calculation	Discr (%)
PRZ pressure [bar]	155.5	155.3	0.129
Primary flow rate [kg/s]	4700	4720	0.426
Core bypass flow rate [kg/s]	199.8	203	1.602
Inlet core temperature [K]	566	562.7	0.583
Outlet core temperature [K]	603	599.6	0.564
Primary coolant volume [m3]	455	463.81	1.936
Core power [MW]	1000	1000	0.000
SG outlet pressure [bar]	58.3	58.9	1.029
SG inlet temperature [K]	497	496.8	0.040
SG outlet temperature [K]	590	593	0.508
Total SG mass flow rate [kg/s]	500	500.45	0.467

Table 2
Sequence of main transient events.

Main event	Signal	Time (s)	Actuation
Double guillotine break opening	\	0	
High Containment Pressure (HCP) set point	HCP signal, SCRAM-Signal	27	SCRAM; Secondary syst. isolation; EHRS
Low Pressurizer Level (LPL) set point	LPL signal	100	RCPs coast-down
	(HCP + LPL)	104	Shroud RI-DC valves open
Low Pressurizer Pressure (LPP) set point	LPP signal	109	
	LM signal (HCP + LPP)	114 121	EBT line opening ADS stage-1 opening
EBT-A emptying	\	416	
Containment maximum pressure reached	\	1590	
Low Differential Pressure RV-Containment (LDPC)	LDPC signal	2115	LGMS-B starts to inject
First DW-Primary circuit pressure equalization	\	2200	
First Reverse flow PSS-DW	\	2535	
Second Reverse flow PSS-DW	\	4655	
EBT-B emptying	\	3150	
Low LGMS mass set-point	Low LGMS signal	13,620	ADS stage 2
Pressure equalization RV-Containment	\	13,860	

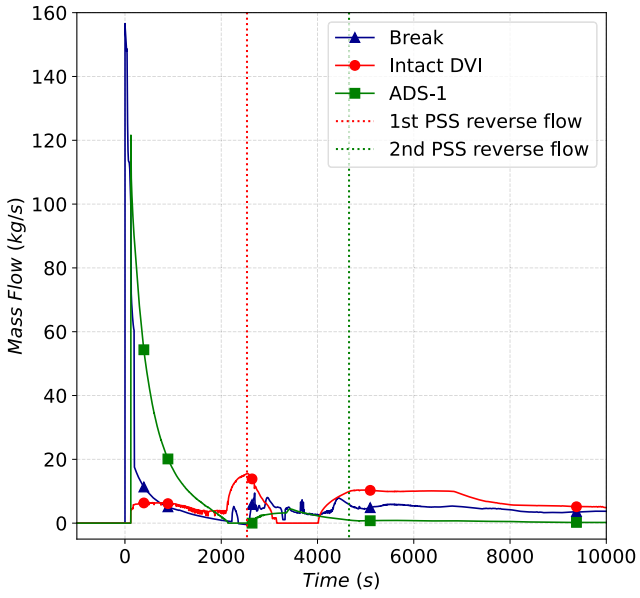


Fig. 4. Mass flow rate through the ADS-1, broken DVI line and intact DVI line (window).

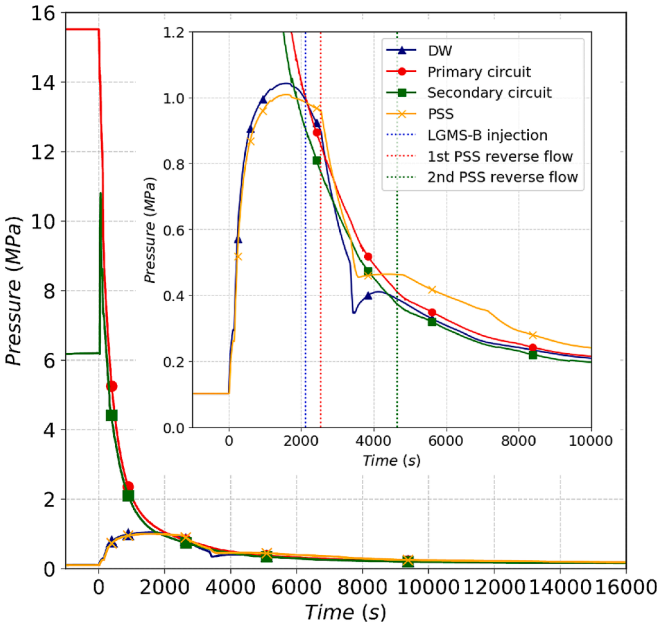


Fig. 5. Pressure evolution in PRZ, secondary circuit DW and PSS.

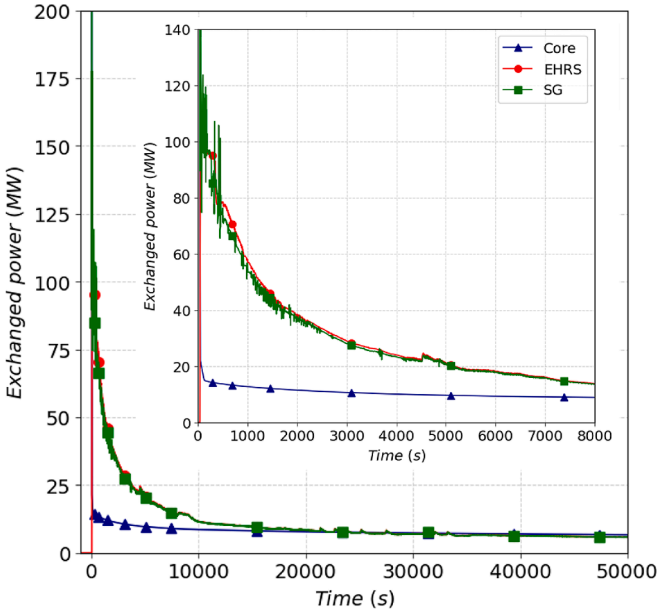


Fig. 6. Decay heat, SGs power from RPV to EHRS loops, power from EHRS to RWST.

The low pressurizer pressure set-point is reached at 109 s, triggering the LM signal that opens the EBT and the ADS-1 valves. The ADS-1 helps to depressurize the RPV to reach the pressure equalization with the DW, by dumping steam and water in the QT. The SPARC model is activated at the ADS-1 connection with the QT. The driver for bubble collapse in the SPARC model is the product between the bubble rise efficiency and the water temperature efficiency. The former accounts for bubble rise distance, the latter for water pool subcooling (Humphries et al., 2024). This system is able to condense steam by liquid-steam direct interaction for the initial transients' phase. However, due to its relatively small volume, it reaches its saturation temperature rapidly, nullifying the water temperature efficiency, leading to smaller condensation. Moving forward, the EBTs start to inject cool water through the DVI lines. The one connected to the broken DVI rapidly discharges the whole inventory in the first 300 s due to the presence of the break. During this stage the water injected by the EBT is sufficient to prevent the heat up of the core, while the increase of the DW pressure is limited by the PSS, which follows the same pressurization trend, together with the LGMS tanks. The condensed steam entering the tanks causes a rise in the gas and liquid temperatures and a slight increase in the water level caused by the condensed steam dumped in the pools. In PhW 1, the primary depressurizes quickly while the DW reaches its peak pressure of 10.4 MPa after 1590 s from the rupture.

PhW 2 starts when the LDPC (low differential pressure between the vessel and the drywell) signal is triggered at 2115 s, and the LGMS valves open, injecting water through the DVI lines. The LGMS tank connected to the broken DVI (LGMS-A) discharges directly to the RC. After an

initial decrease of the water level inside the core during the first 1800 s, the intact DVI provides enough make-up water to the vessel through the LGMS being able to restore the water levels inside the core, eventually completely reflooding it at 7000 s as seen in Fig. 1, Fig. 7 and Fig. 9. During PhW 2 the core power is removed by the EHRS and the containment is cooled through the DW walls and indirectly by the EHRS (cold water ejection inside the DW). As the pressure difference between the DW and the primary circuit is eventually nullified and reverted at 2115 s, the LGMS-B injection starts (Fig. 5). Right before the aforementioned injection starts, we notice a slow decreasing pressure trend in the DW and PSS. The DW atmosphere pressure is decreasing due to two effects: containment cooling and heat exchange between the superheated steam in the DW and subcooled water in the cavity, highlighted in Fig. 10. On the other hand, the PSS system is slowly depressurizing due to LGMS injection. The PSS system pressure does not decrease as fast as the DW one, although they are hydraulically connected. This is due to the previously pressurized atmosphere inside the PSS, which indeed forces the PSS vent's water level to rise (also countering the pressure decrease in this system by increasing the vent's head) until it reaches its maximum height as depicted in Fig. 11. Once the vent's level reaches its maximum height, the first PSS reverse flow occurs, which consists in a water discharge from the PSS to the DW. This decreases the pressure in both the DW, and the PSS as shown in Fig. 5. This reverse flow stops at around 3600 s when the driving forces in play (pressure difference between PSS and DW and PSS vent gravity head) are equalized and reverted (the vent line level decreases in Fig. 11). Now the Drywell pressure increases again, following the primary circuit pressure. While this happens, the PSS remains pressurized and since both the DW and the primary pressure are decreased due to the cooling effects previously described, a second reverse flow is triggered (the vent's level rises again to its maximum at 4655 s). This second reverse flow is stopped at 6900 s for the same reasons as the first one. There are no successive reverse flows, since the PSS water level and the driving pressure difference are not enough to cause further injections. This phenomenology is illustrated for the PSS-B system and is basically identical for the PSS-A system. It is interesting to highlight that the main pressure contributor for the DW volume is steam, while it is Nitrogen for the PSS system. Indeed, the partial pressure of steam in DW is almost equal to the curve shown in Fig. 5 for the DW, while the partial pressure of Nitrogen in the PSS is identical to the PSS curve. This shows that the steam entering the PSS

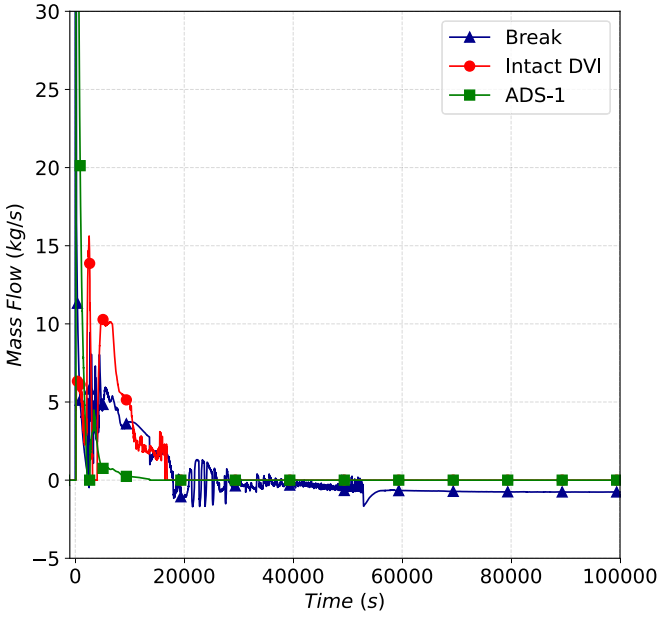


Fig. 7. Mass flow rate through the ADS-1 broken DVI line and the intact DVI line (long term).

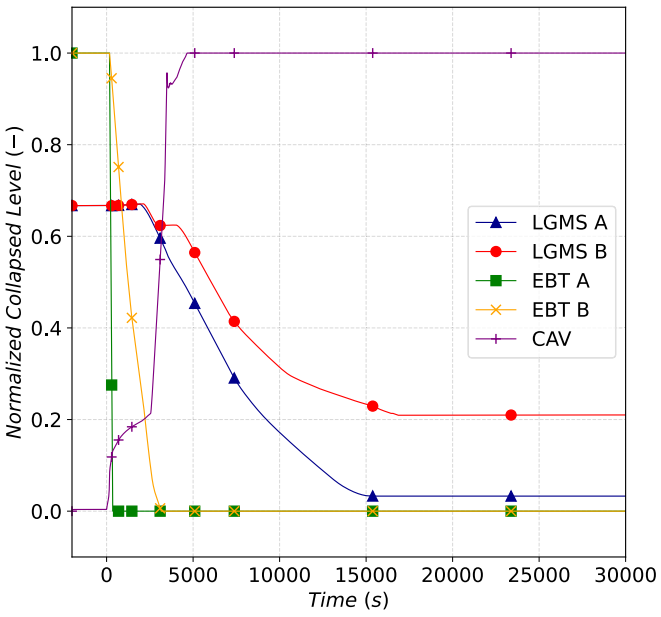


Fig. 8. Normalized water level inside the tanks of LGMS (A and B), EBT (A and B) and reactor cavity.

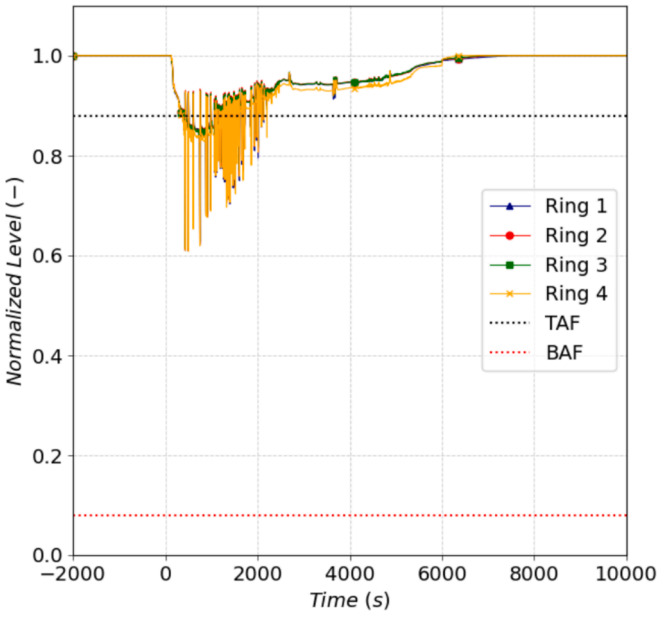


Fig. 9. Normalized levels inside the core channels (between upper and lower nozzle grids).

system is indeed condensing by liquid-steam interactions, since the SPARC model is activated in the flow paths connecting the PSS to the DW. The large water inventory inside the PSS tanks makes the RC flood rapidly, as shown in Fig. 8. Since the cavity water reaches the break's elevation, a reverse flow is possible in the long term as the pressure transient is over. The ADS- stage 2 actuation results in equalizing pressures, ending PhW2 (13620 s).

The first reverse mass flow rate through the vents can reaches 130 kg/s, with a short peak in its final moments, while the second ejects around 40 kg/s for its whole duration. The LGMS tank connected to the broken DVI line empties faster than the one injecting make-up water to the RPV reaching the Low LGMS signal (LGMS water level signal). Fig. 8 depicts the normalized level inside the LGMS, EBT and RC. The passive

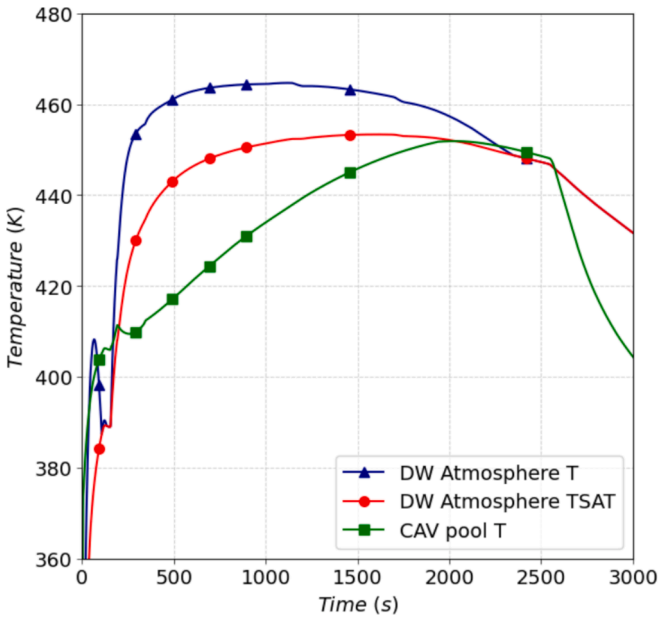


Fig. 10. Cavity pool temperature, DW atmosphere Temperature and saturation temperature trends.

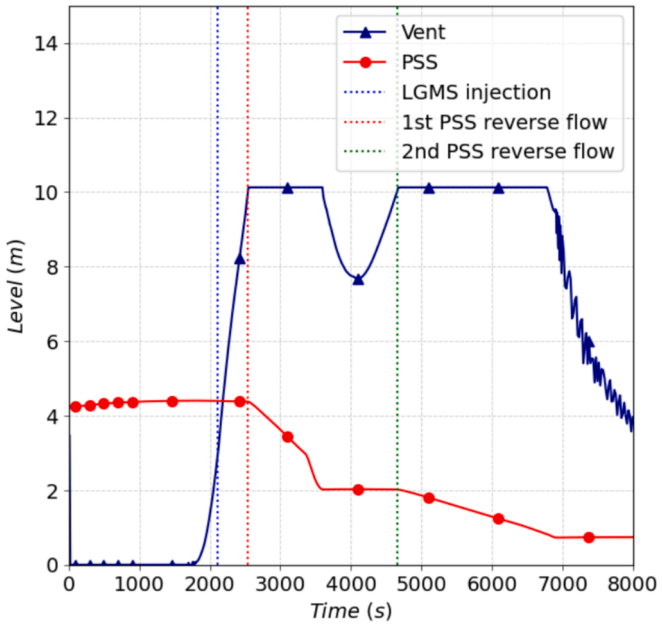


Fig. 11. PSS vent line and tank levels during PhW 2.

safety systems guarantee the coolant reintegration of the primary circuit, which allows the natural circulation inside the RPV to take place, flooding the uncovered core (Fig. 9). The Top of Active Fuel remains uncovered for less than 1000 s, however no cladding damage, no oxidation or hydrogen production is registered.

Fig. 12 shows the natural circulation through the RI-DC check valves. The natural circulation inside the RPV is characterized after the accident by a two-phase flow, with a reduction of the quality during the evolution of the transient. Oscillations are observed in the SG removed power (Fig. 6), and in the mass flow established both in the primary circuit (Fig. 12) and the EHRS. Oscillating behaviour is observed as in (Bianchi and Ferri).

The PhW 3 or cool down phase is characterized by the RWST water heat up (until saturation temperature), the reduction of the temperature

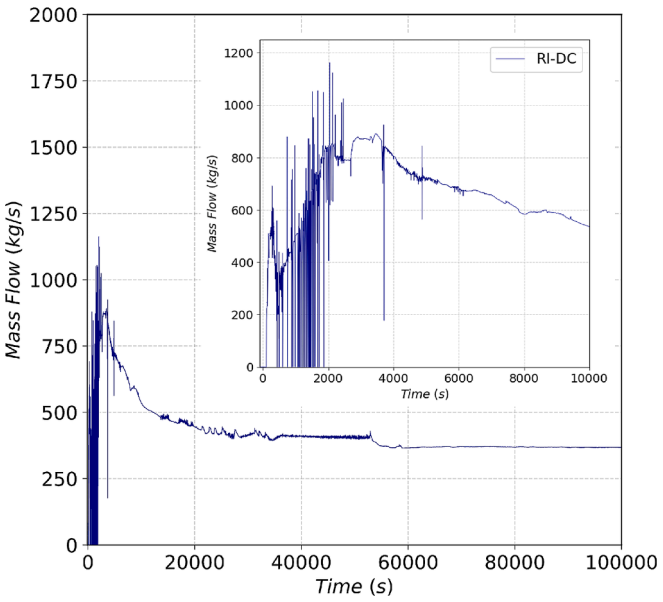


Fig. 12. Mass flow rate though the RI-DC check valves.

gap between the EHRS HL and the pool water leads to the reduction of heat removal, which becomes almost the same as the core decay heat. Once the water levels of the LGMS-A tank reach the Low LGMS water level set-point, the ADS-2 valve opens, and the equilibrium between the primary and the DW is set, following the same depressurization trend. The saturation temperature in the RWST is reached at about 110000 s (30.5 h), the thermal stratification inside the RWST is flattened, while natural circulation in the pool takes place for the full duration of the transient. Axial temperatures inside the RWST volumes are shown in Fig. 14, in which volumes are enumerated based on their height: 1 corresponds to the bottom of the pool, 8 to the top (refer to Fig. 2 for visualization).

In this phase, the core is kept covered thanks to the break connection to the filled RC, and the temperature inside the core is kept almost constant at less than 400 K (Fig. 13). The long-term cooling is capable to remove the decay heat produced in the core by the primary side natural circulation that, through the SGs transfers energy to the EHRS and the

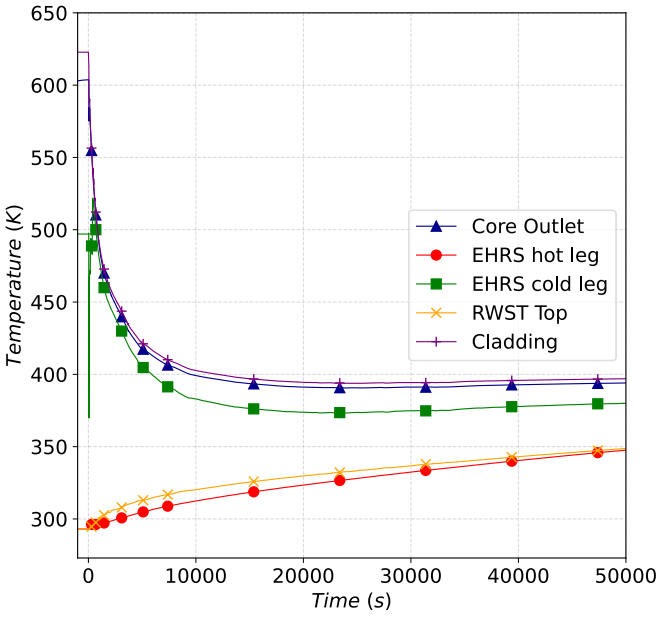


Fig. 13. Temperature of cladding, EHRS hot leg, outlet of core and RWST.

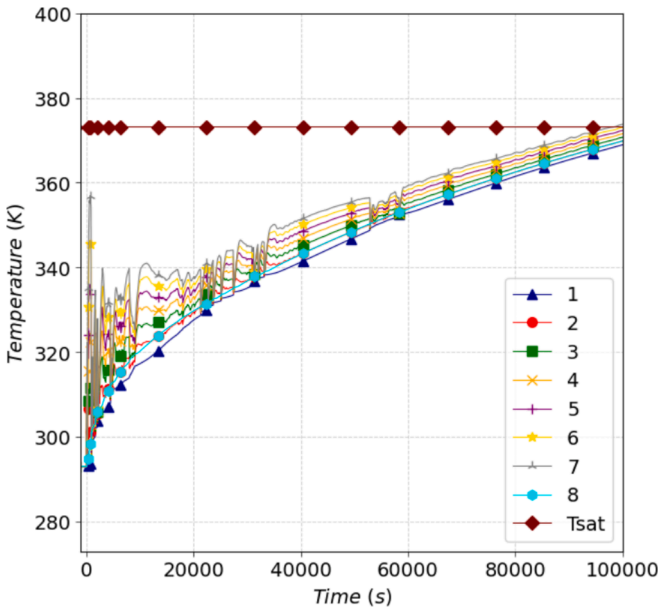


Fig. 14. RWST axial temperature profiles.

heat losses from the containment walls to the environment. The reactor reaches an almost steady-state regime at the end of the PhW 3. Due to a slight imbalance between EHRS removed power and Decay power, a slow increase transient is observed in the long term (up to 53 h), for the DW and the core temperature. The former settles at around 3.0e5 Pa, while the latter at 407 K. Between 53 to 72 h, the system finds itself in a cold safe steady state.

The simulation has shown important insights on the foreseen phenomena of the safety systems analyzed during the transient simulation carried out with MELCOR, as summarized in Table 3. The passive safety systems performances are in accordance with the magnitude of expected results (Maccari et al., 2021); (Achilli et al., 2012) and at the end of the simulation no core damage occurred. The safety systems were able to remove the decay heat by the EHRS, and the steam condensed on the DW walls was dumped into the RC, leading the reactor to a safe shutdown state as expected. The RC plays an important role regarding the flooding of core, providing a long-term gravity makeup.

Table 3
Foreseen phenomena by MELCOR simulation.

System/ Component	Foreseen phenomena by MELCOR simulation	Reference Figure/ Table
RPV	Single and two-phase natural circulation	Fig. 10
	DVI break critical flow	Fig. 4, Fig. 7
	Heat transfer in SG primary side	Fig. 6, Fig. 9, Table 1
	Heat transfer in covered and partially uncovered core	
SG	Heat transfer in SG secondary side	Fig. 6, Table 1
	Liquid Boiling in SG Helical Coil	
	Steam superheated in secondary side	
EHRS	Steam condensation in heat exchanger (heat transfer in tubes side)	Fig. 6, Fig. 13
	High and low pressure heat exchange in exchangers	
	Two phase natural circulation	
RWST	Natural convection in the pool	Fig. 6, Fig. 14
	Thermal stratification	
	Heat transfer pool side	
EBT	Gravity-driven injection	Fig. 4, Fig. 7, Fig. 8
LGMS	Gravity and differential pressure driven injection	Fig. 4, Fig. 7, Fig. 8, Fig. 5
PSS	Direct condensation for liquid-steam interaction	Described below Fig. 8 and Fig. 9
DW	Condensation on containment	Text below Table 2, Fig. 5
	Effect of non-condensable gas on condensation heat transfer	
	Heat losses to the environment	
	Primary/containment coupling	
RC	Gravity-driven injection	Fig. 7
Quench Tank	Direct Condensation for Liquid-Steam interaction	Described below Fig. 8

5. Conclusions

The present iPWR accident analysis, developed in the SASPAM-SA Horizon Euratom project, provided interesting information about the MELCOR integral code capability to simulate the associated DBA transient phenomena in a generic iPWR using a passive mitigation strategy, as starting point for future postulated plausible BDBA and severe accident sequences studies. The overall accident evolution is in accordance with the expected plant behavior, and the phenomenological windows are reproduced.

The MELCOR model was able to reproduce the reference steady-state conditions. Analyzing the simulation results of the DBA transient it can be concluded that the code can simulate the thermal–hydraulic coupling between containment, the RPV, and passive safety systems, which permits the passive mitigation strategy of the reactor, leading to the RPV depressurization and core cooling in safety conditions. Also, the specific phenomena that take place in the different passive systems are predicted (e.g. steam condensation in heat exchanger, RWST natural convection, etc.). The phenomenology predicted by MELCOR is consistent with other studies presented in public literature. In order to assess the result quantitatively, a validation activity against selected phenomena is necessary. Due to the importance of the passive decay heat removal system possible assessment studies of interest could be the assessment of the MELCOR code capability against all the PERSEO experimental campaign (Bandini et al., 2011); further assessment studies that characterize the capability of the MELCOR code to reproduce natural circulation at different core powers; an example could be the application of MELCOR code against the DOE-OSU MASLWR-002 and IAEA_ICSP test 3 (Mascari et al., 2016); (International Atomic Energy Agency, 2013). This will allow also to assess quantitatively the code capability to predict the SG heat transfer from primary to secondary circuit with helical coil steam generators.

From the analysis, we can summarize the following conclusions. In terms of scenarios investigated, the prediction of the EHRS, PSS and containment behavior is fundamental in the overall evolution of the plant. This is due to the strong interaction between the containment and the integral RPV, which depends on the activation of the safety systems. The actuation of both the ADS lines is fundamental during the accident scenario, allowing the primary circuit fast depressurization, containment and RPV equalization and the maintenance of the coolant inside the RPV. Further derivative activities focused on assessing the capability of the MELCOR code to simulate the phenomena taking place in postulated plausible severe accident scenarios in iPWR are planned inside SASPAM-SA.

CRediT authorship contribution statement

M. Principato: Writing – review & editing, Software, Investigation. **F. Giannetti:** Writing – original draft, Methodology, Investigation, Conceptualization. **M. Imperatori:** Investigation. **M. D'Onorio:** Writing – review & editing, Conceptualization. **M. Garcia:** Validation. **L.E. Herranz:** Formal analysis. **A. Bersano:** Methodology, Conceptualization. **F. Mascari:** Writing – review & editing, Supervision, Project administration, Funding acquisition.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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Data availability

Data will be made available on request.

References

Achilli, A., Congiu, C., Ferri, R., Bianchi, F., Meloni, P., Grgić, D., Dzodzo, M., 2012. SPES3 Facility RELAP5 sensitivity analyses on the containment system for design review, Sci. Technol. Nucl. Install. **2012**, art. n. 173637.

AP300 SMR, Westinghouse website, available online on https://westinghousenuclear.com/energy-systems/ap300-smr/.

Bandini, G., Meloni, P., Polidori, M., Lombardo, C., 2011. Validation of CATHARE V2.5 thermal-hydraulic code against full-scale PERSEO tests for decay heat removal in LWRs. Nucl. Eng. Des. 241 (2011), 4662–4671.

Beeny, B., Humphries, L., Luxat, D., Schmidt, R., 2020. MELCOR for SMRs: Modeling ad Industry/Regulatory Use, Sandia National Laboratories (SNL), US NRC, CNL, CNSC & AECL Meeting Severe Accident Research and SMRs, USA, March 2 - February 7, Available online on https://www.osti.gov/servlets/purl/1769572.

Bianchi, F., Ferri, R., SPES-IRIS Facility RELAP5 sensitivity analyses on the containment system for design review, SIET technical report RdS/2010/61 available on ENEA-IRIS open archive.

Carelli, M.D., Conway, L.E., Oriani, L., Petrović, B., Lombardi, C.V., Ricotti, M.E., Barroso, A.C.O., Collado, J.M., Cinotti, L., Todreas, N.E., Grgić, D., Moraes, M.M., Boroughs, R.D., Ninokata, H., Ingersoll, D.T., Oriolo, F., 2004. The design and safety features of the IRIS reactor. Nucl. Eng. Des. 230, 151–167.

Esmaili, H., 2018. Independent Assessment of In-Vessel Retention and Steam Explosion for the NuScale Small Modular Reactor, RES/FSCB 2018.02, USNRC.

European Commission, Safety Analysis of SMR with PAssive Mitigation strategies - Severe Accident - SASPAM-SA Project, 2023. Available online on https://cordis.europa.eu/project/id/101059853.

European SMR pre-Partnership Reports Workstream 5 –Research, Development, and Innovation Roadmap, https://www.nuclearеurope.eu/project/european-smr-pre-partnership/.

Fakhraei, A., et al., 2020. Safety analysis of an advanced passively-cooled small modular reactor during station blackout scenarios and normal operation with RELAP5/SCDAP. Ann. Nucl. Energy 143, 107470.

Ferri, R., Congiu, C., 2008. Conceptual design of the SPES3-IRIS facility, SIET 01 334 RT 07 Rev. 1, Piacenza (Italy).

Francis, D., et al., 2024. NUWARD SMR safety approach and licensing objectives for international deployment. Nucl. Eng. Technol. 56 (3), 1029–1036.

Horizon Euratom Safety Analysis of SMR with PAssive Mitigation strategies - Severe Accident (SASPAM-SA), available at https://zenodo.org/communities/saspam-sa/about.

Humphries, L.L., Beeny, B.A., Gelbard, F., Louie, D.L., Phillips, J., 2024. MELCOR Computer – Code Manuals, Vol 1: User Manual, Version 2024.0, Sandia National Laboratories, Albuquerque (USA).

Humphries, L.L., Beeny, B.A., Gelbard, F., Louie, D.L., Phillips, J., 2024. MELCOR Computer – Code Manuals, Vol 2: Reference Manual, Version 2024.0, Sandia National Laboratories, Albuquerque (USA).

IAEA, 2022. Advances in small modular reactor technology developments, 2022 Edition, in IAEA advanced reactor information system (ARIS), Vienna.

International Atomic Energy Agency, 2013. Evaluation of Advanced Thermohydraulic System Codes for Design and Safety Analysis of Integral Type Reactors, IAEA-TECDOC-1733. IAEA, Vienna.

IRIS International Reactor Innovative and Secure, IRIS Plant Overview, 2002. Available online on https://www.nrc.gov/docs/ML0336/ML033600086.pdf.

Jiang, N., et al., 2019. Thermal margin evaluation for the stratified molten pool in SMR-IP200. Nucl. Eng. Des. 355, 110354.

Maccari, P., Mascari, F., Ederli, S., Meloni, P., Manservisi, S., 2021. Astec code DBA analysis of a passive mitigation strategy on a generic IRIS SMR. Ann. Nucl. Energy 156, 108194.

Malicki, M., et al., 2019. Development of MELCOR thermal hydraulic model of AP1000 and its verification for a DECL break. Ann. Nucl. Energy 128, 44–52.

Malicki, M., Lind, T., 2024. Melcor 2.2 iPWR LOCA type accident analysis, part I: thermal-hydraulics. Nucl. Eng. Des. 428, 113433.

Mascari, F., 2022. Safety Analysis of SMR with Passive Mitigation strategies – Severe Accident (SASPAM-SA) Horizon Euratom Project, ETSON NEWS, September, Available online on https://www.etson.eu/node/290.

Mascari, F., et al., 2023. OECD/NEA/CSNI/WGAMA PERSEO benchmark: Main outcomes and conclusions. Nucl. Eng. Des. 405, 112220. https://doi.org/10.1016/j.nucengdes.2023.112220.

Mascari, F., Nakamura, H., Umminger, K., De Rosa, F., D'Auria, F., 2015. Scaling issues for the experimental characterization of reactor coolant system in integral test facilities and role of system code as extrapolation tool. Proceedings of the 16th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-16), Chicago, IL, August 30-September 4, 2015. University of Pisa. Retrieved from https://arpi.unipi.it/retrieve/handle/11568/835858/177689/1008-13916nostri.pd.

Mascari, F., De Rosa, F., Woods, B.G., Welter, K., Vella, G., D'Auria, F., 2016. Analysis of the OSU-MASLWR 001 and 002 tests by using the TRACE code (NUREG/IA-0466). U.S. Nuclear Regulatory Commission. Retrieved from https://www.nrc.gov/reading-rm/doc-collections/nuregs/agreement/ia0466/index.html.

Mascari, F., Woods, B.G., Welter, K., D'Auria, F., Bersano, A., Maccari, P., 2023. Small modular reactors and insights on passive mitigation strategy modeling. Nucl. Eng. Des. 401, 112088.

Mascari, F., Bersano, A., Woods, B.G., Reyes, J.N., Welter, K., Nakamura, H., 2024. Scaling-up assessment of natural circulation phenomena in integral Small Modular Reactor by TRACE code. Nucl. Eng. Des. 420, 113018. https://doi.org/10.1016/j.nucengdes.2024.113018.

Matsuura, M., 2023. Hitachi-GE's Activities for Development of Small Modular Reactor BWRX-300, the 30th International Conference on Nuclear Engineering (ICONE-30), Kyoto, Japan.

Nuclear Regulatory Commission, 2014. MELCOR Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project, NUREG/CR-7008, Albuquerque (USA).

Park, R., et al., 2021. Development of IVR-ERVC evaluation method and its application to the SMART. Ann. Nucl. Energy 161, 108463.

SASPAM-SA website, available at https://www.saspam-sa.eu.

SASPAM-SA, 2023. Milestone 2.1: Generic Data on Design 1 and Design 2.

Seyffert, C., Malicki, M., 2024. Validation methods in modelling the PANDA IPSS experiment: a MELCOR 2.2 assessment of a passive isolation condenser. Prog. Nucl. Energy 177, 105430.

SMR Regulators' Forum Pilot Project Report: Considering the Application of a Graded Approach, Defence-in-Depth and Emergency Planning Zone Size for Small Modular Reactors, January 2018 https://www.iaea.org/sites/default/files/18/01/smr-rf-report-29012018.pdf.

Włostowsky, M., Darnowski, P., 2024. Study of the DVI-LOCA in the AP1000-like reactor with MELCOR code. Ann. Nucl. Energy 200, 110397.

Yoon, D.S., Jo, H.J., Corradini, M.L., 2017. Assessment of MELCOR condensation models with the presence of noncondensable gas in natural convection flow regime. Nucl. Eng. Des. 317, 110–117. https://doi.org/10.1016/j.nucengdes.2017.03.018.

Zhu, D., et al., 2016. Evaluation of in-vessel corium retention margin for small modular reactor ACP100. Ann. Nucl. Energy 94, 684–690.

Zukasukas, A., 1972. Heat transfer from tubes in crossflow. Adv. Heat Transf. 8, 93–160.