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Analysis of BDBA sequences in a generic IRIS reactor using ASTEC code

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ABSTRACT

Integral Severe Accident (SA) codes are aimed at providing an exhaustive coverage of all the main phenomena taking place in a core melt accident. Today, these deterministic codes have reached a high level of maturity for the simulation of operating reactors and the nuclear technical community is starting to extend their applicability to advanced reactor designs, as Small Modular Reactor (SMR). In the framework of the NUGENIA TA-2 ASCOM (ASTEC COMmunity) collaborative project, a generic input-deck based on the IRIS design has been developed for the ASTEC code. The generic SMR ASTEC model has been already proved able to simulate the main thermal–hydraulic phenomena driving the passive mitigation of a SBLOCA in Design Basis Accident (DBA) conditions. The same initiator event, regarding the guillotine break of a Direct Vessel Injection (DVI) line, will be assumed for the simulation of beyond-design scenarios by considering the unavailability of selected passive safety systems. The results of the ASTEC simulations of four Beyond Design Basis Accidents (BDBAs) (study carried out with ASTEC V2.2, IRSN all rights reserved, [2021]) will be analyzed and discussed against the reference DBA sequence in the present paper. This study is aimed at proving the first insights about the capability of the ASTEC model of a generic IRIS reactor to be used in BDBA and in SA analyses, if significant core degradation takes place. In addition, it characterizes the role played by each safety system in SMR passive mitigation strategy and give the possibility to characterize the phenomenologies specific of SMR designs.

1. Introduction

In the last decades, the international nuclear technical community started to develop advanced reactor designs that, starting from the experience on operating Light Water (LW) Nuclear Power Plants (NPPs), aim to improve safety features, to increase economic efficiency and to reduce capital costs (Mascari et al., 2020). Many new advanced reactors design adopt passive mitigation strategies based on natural-driven forces for the removal of the core power during accidental conditions, or even in nominal operation (IAEA-TECDOC-, 2004; Mascari, 2010). In this framework, advanced Small Modular Reactors (SMRs), using passive safety systems, are particularly interesting because of their potential economic and safety advantages (Iaea, 2014). SMRs are usually classified as reactors having a maximum power output of 300 MWe (Iaea, 2014) and, in relation to the safety, it implies a lower decay power to be removed in postulated accidental scenarios. LW SMRs, using passive mitigation strategies, are in general characterized by common features with the current operating large-LWR and by other specific features typical of their evolutionary designs, providing inherent safety advantages that reinforce the first three levels of the Defence-in-Depth (DiD) principle. In this view, integral Severe Accident (SA) codes, such as ASTEC (Chatelard et al., 2016; Chatelard et al., 2014), MELCOR (MELCOR, 2022), etc., developed for the simulation of conventional-size reactors, have reached a high level of maturity and, today, the international nuclear technical community is starting to analyze their applicability to advanced designs. In particular, in order to apply a SA code for the design of accident management's strategy in SMR or for a safety review process, the code simulation capabilities should be validated for integral configurations, compact containments and the passive mitigation strategies. In the framework of the NUGENIA TA-2 ASCOM (ASTEC COMmunity) collaborative project (Chatelard, 2018; NUGENIA, 2022), coordinated by IRSN, a generic input-deck based on the IRIS (International Reactor Innovative and Secure) design (IRIS International Reactor Innovative and Secure and Plant Overview, 2002; IRIS Plant Description Document, 2003) has been developed with the SA code ASTEC V2. The generic IRIS nodalization was proved able to simulate the main thermal–hydraulic phenomena driving the passive mitigation of a Small Break Loss of Coolant Accident (SBLOCA) in DBA conditions (Maccari et al., 2021). The application of a SA code to simulate a SMR

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Nomenclature

ADS	Automatic Depressurization System
ASTEC	Accident Source Term Evaluation Code
BDBA	Beyond Design Basis Accident
CRDM	Control Rod Drive Mechanism
DBA	Design Basis Accident
DC	Downcomer
DiD	Defence-in-Depth
DP	Differential Pressure
DVI	Direct Vessel Injection
DW	Drywell
EBT	Emergency Boration Tank
EHRS	Emergency Heat Removal System
IRIS	International Reactor Innovative and Secure
IRSN	Institut de Radioprotection et de Sûreté Nucléaire
IVMR	In-Vessel Melt Retention
LBLOCA	Large Break Loss Of Coolant Accident
LGMS	Long-term Gravity Make-up System
LH	Lower Head
LP	Lower Plenum
LW	Light Water
NPP	Nuclear Power Plant
PCS	Primary Coolant System
PRZ	Pressurizer
PSS	Pressure Suppression System
PWR	Pressurized Water Reactor
QT	Quench Tank
RC	Reactor Cavity
RPV	Reactor Pressure Vessel
RWST	Refueling Water Storage Tank
SA	Severe Accident
SBLOCA	Small Break Loss of Coolant Accident
SCS	Secondary Cooling System
SG	Steam Generator
SMR	Small Modular Reactor
SOT	Start Of the Transient
TAF	Top of Active Fuel.

design in BDBA and core degradation conditions could lead to valuable conclusions regarding the capability of the code to simulate the main thermal–hydraulic and core degradation phenomena in integral configuration and in smaller containment (characterized by a reduced containment-volume/core-power ratio compared to large-LWR). Besides this purpose, the simulation of BDBAs add valuable information regarding the role played by each passive system in the mitigation strategy with respect to the DBA sequence. In the present work, the generic ASTEC model of the IRIS reactor has been used for the development of four BDBA scenarios. The four scenarios have been postulated in view of providing the first insights about the capability of the ASTEC model of the generic IRIS to be used in BDBA and in SA analyses, to characterizes the role played by each safety system in a SMR passive mitigation strategy and to characterize the phenomenology specific of SMR designs in SA conditions. The simulations results have been analyzed and discussed against the reference DBA simulation. Similar studies to the present activity can be found in (Di Giuli et al., 2015; Li et al., 2017; Skolik et al., 2021).

2. Description of iris-like reactor design

IRIS is an integral, modular, medium power (300 MWe) Pressurized Water Reactor (PWR), developed by an international consortium led by Westinghouse and involving several Universities, companies and organizations (Carelli et al., 2009; Ferri and Congiu, 2008). The SMR design consists of a Reactor Pressure Vessel (RPV) which includes all the reactor Primary Coolant System (PCS) components: the reactor core, the Pressurizer (PRZ), Control Rod Drive Mechanism (CRDM), Steam Generators (SG), primary coolant pumps. The integral arrangement of the PCS layout avoids high pressure components outside the RPV and large primary vessel penetration, eliminating the possibility of Large Break LOCAs (LBLOCAs) and reducing the number of possible SBLOCAs (safety by design concept) (Ferri et al., 2012; Achilli et al., 2012). As shown in Fig. 1, the reactor core is located in the bottom part of the RPV and the PRZ is integrated in the upper head. The primary coolant coming from the core outlet, flows upward through the inner riser channel and reaches the top part of the circuit. The RPV of IRIS is placed in a spherical containment, as shown in Fig. 2, in which most of the passive safety systems are located. The spherical containment is itself part of the passive safety approach, and in case of LOCA is directly involved in the mitigation strategy (Ferri et al., 2012). The reactor passive safety systems includes: an Emergency Heat Removal System (EHRS) consisting of 4 independent trains working in natural circulation, each of which is

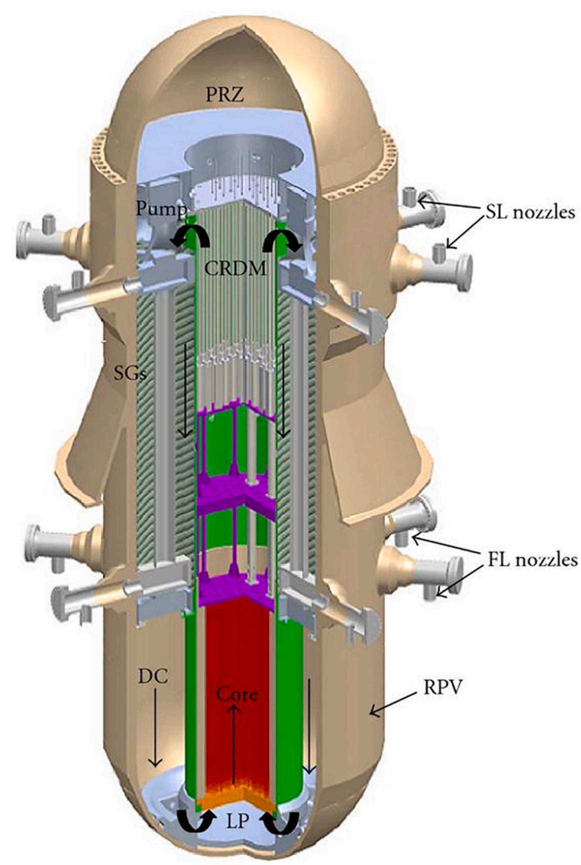

Fig. 1. IRIS integral reactor vessel layout (IRIS International Reactor Innovative and Secure and Plant Overview, 2002).

connected to a pair of the 8 SGs; 3 trains of an Automatic Depressurization System (ADS), on the head of the RPV, dumping steam in the pool of the same Quench Tank (QT) which is located in the Drywell (DW); 2 Emergency Boration Tanks (EBTs), to inject water for gravity; two bigger tanks as Long term Gravity Make-up System (LGMS); 2 Pressure Suppression System (PSS) tanks connected to the DW through a vent pipe. The PSS are the largest tanks and besides limiting the

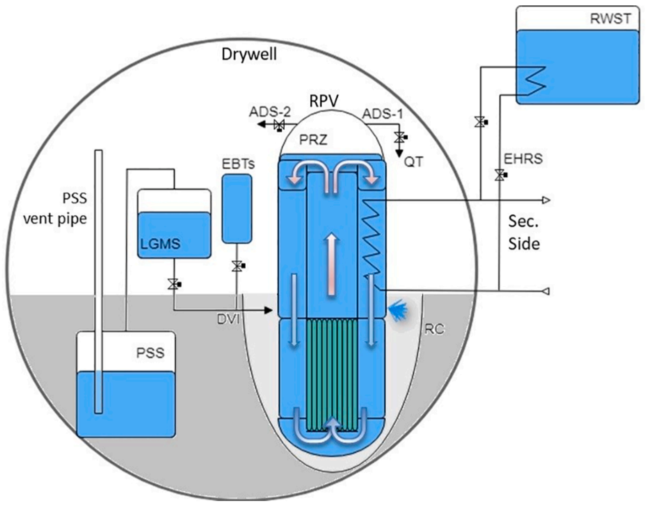

Fig. 2. Scheme safety systems for a generic IRIS reactor (for the sake of simplicity, only one line is reported for redundant systems) (Maccari et al., 2021).

containment pressurization, this system is also employed to fill the Reactor Cavity (RC) of water in the mitigation strategy. Each of the 2 EBT, LGMS, and PSS tanks belongs to one of the 2 independent lines, as the one reported in Fig. 2. The PSS tank and the LGMS tank, belonging to the same line, are connected through an open vent pipe which keeps in communication the top parts of the two tanks, maintaining the two tanks at the same pressure (see Fig. 2). The EBTs and LGMS tanks on the same line can inject cooling water in the vessel through a Direct Vessel Injection lines (DVIs) once the respective valves are opened. The energy removed by natural circulation in the EHRS is transferred to two Refueling Water Storage Tanks (RWSTs), acting as heat sink and located outside the containment.

2.1. IRIS passive mitigation strategy for a DVI line double ended break

The DBA sequence selected for the presented study is a double-ended break of one of the two DVI lines (2-inches of diameter), considering the availability of all the emergency passive safety systems. The SBLOCA considered is the largest break at the lowest elevation that may cause a LOCA in the IRIS plant. The failure is located on the primary circuit, above the Top of Active Fuel (TAF) (Achilli et al., 2012; Bianchi and Ferri, 2010). The passive mitigation strategy consists in the opening of specific safety valves which activate specific safety systems, consequently to the triggering of set-point signals. The DVI line-A is considered as the broken line and the DVI line-B as the intact one.

The main phenomenology of the DBA sequence is described in the following according to (Carelli et al., 2009; Achilli et al., 2012; Bianchi and Ferri, 2010):

a) The accident initiator event is the guillotine break of the DVI line-A; it is followed by the RPV blowdown and depressurization. The containment pressure starts to increase and causes the transfer of a hot steam-gas mixture from DW to PSS tanks through the PSS vent lines. It determines the steam condensation underwater, limiting the DW pressurization. The PSS and LGMS tanks pressure increases driven by the containment pressure trend. The RC is slowly filled by the primary water flowing from the break and by the steam condensation on the DW metal surfaces.
b) When the containment pressure value reaches the set-point, the S-signal (Safety) triggers the reactor SCRAM, the secondary-side lines isolation and the opening of two of the four EHRS loops. Due to the SCS isolation, the SCS pressure rapidly increases to the saturation point. Natural circulation starts in the EHRS loops 1 and 2, with the transfer of power to the RWST water, contributing to the RPV cooling and depressurization.
c) The primary pumps coast down is activated by the low PRZ level signal. The decrease of pump delivery pressure determines the automatically opening of the RI-DC check valves, increasing the core cooling by only natural circulation.
d) When the low PRZ pressure set-point is reached, the LM-signal (LOCA Mitigation) occurs actuating the ADS stage-1, the EBTs and the two EHRS remaining loops. Due to the EBT valves opening, cold borated water starts to be injected through the two DVI lines by gravity, while the broken loop (line-A) drops the EBT water inside the RC. The ADS stage-1 discharges steam from the PRZ head to a QT, increasing the PRZ depressurization rate and the equalization between RPV and containment pressures. The energy removal from the PCS is increased by the actuation of all the EHRS loops.
e) The low RPV-Containment Differential Pressure (DP) signal, activated by the RPV and DW pressures equalization, determining the opening of the valves connecting the LGMS line-B to the RPV (through the intact DVI line) and the LGMS line-A to the RC (through the broken DVI line). Steam condensation on the containment walls and heat removal from primary side due to the EHRS intervention causes the containment and the RPV pressure decreases. When DW pressure decreases below PSS pressure, the PSS water is pushed inside the PSS vent pipes to the top end and fills the RC above the break level.
f) When the LGMS mass reaches the low water mass signal, the ADS stage-2 valves are opened permitting steam circulation between RPV and DW. During the long-term cooling phase, the core is kept filled and cooled by the water available from RC, the power removed by the EHRS system and the heat losses to the environment through the DW metal surfaces.

3. Description of ASTEC code and ASTEC nodalization of the generic iris-like reactor

The ASTEC code (Chailan et al., 2017; Laborde, et al., 2021; Chatelard and Laborde, 2022) (Accident Source Term Evaluation Code), developed by the French "Institut de Radioprotection et de Sûreté Nucléaire" (IRSN), aims at simulating an entire Severe Accident (SA) sequence in nuclear water-cooled reactors from the initiating event through the release of radioactive elements out of the containment (Mascari et al., 2019). ASTEC features a modular structure and each module is aimed at simulating a specific set of physical phenomena or related to a specific reactor zone. The main applications of the ASTEC code are source term evaluation studies, Probabilistic Safety Assessment level-2 (PSA2) studies, accident management studies, etc. (Caroli et al., 2015). In the present work, the modules CESAR, CPA, ICARE, SOPHAEROS and ISODOP of ASTEC have been implemented in the generic IRIS reactor model. The CESAR module is dedicated to the simulation of coolant systems thermal-hydraulics, it is a two-phase system code based on a two-fluid 5 or 6-equations thermal–hydraulic model. CPA is the ASTEC module with the purpose of simulating all the relevant thermal–hydraulic processes and plant states taking place in the containment of a LWR. ICARE is the module dedicated to core internals heat-exchange and in-vessel degradation phenomena. ICARE implements mechanical models, processes chemical reactions, incorporates fission product release (by coupling with ELSA) and describes core thermal behavior, degradation and relocation in the Lowe Plenum (LP), until the rupture of the Lower Head (LH) wall. The ISODOP module simulates the decay of Fission Products (FPs) and actinide isotopes and it allows to estimate the decay heat and the isotopes inventory in the different zones of the reactor. SOPHAEROS is in charge of simulating the transport of vapors and aerosols FPs in the RPV and in the containment, accounting for the chemical reactions and speciation. The ASTEC version used is the V2.2 beta, with a 5-equation model for CESAR.

3.1. ASTEC nodalization of the generic IRIS-like reactor

In order to develop the ASTEC nodalization of the generic IRIS-like SMR, no proprietary data have been used; therefore, the main geometric information have been determined by scaling the data available from the SPES-3 facility (Ferri and Congiu, 2008; Ferri et al., 2012; Achilli et al., 2012; Bianchi and Ferri, 2010) or by engineering evaluation. The public general data available for the IRIS reactor can be found in (IRIS International Reactor Innovative and Secure and Plant Overview, 2002; IRIS Plant Description Document, 2003). The approach followed in the generic IRIS modelling with ASTEC V2 aims at accurately simulating the thermal-hydraulics of RPV and passive safety systems. For this reason, CESAR (Fig. 3) has been used for the nodalization of the primary systems (top half of RPV); secondary system (feed and steam lines; SGs) and most of the passive safety systems. The lower half of RPV, including the core, has been modelled with ICARE (Fig. 4).

CPA completes the thermal–hydraulic nodalization modeling the spherical containment (DW and RC). A more detailed description of the thermal–hydraulic nodalization adopted can be found in (Maccari et al., 2021). The reactor meshing considered by SOPHAEROS includes the RPV volumes and all the zones of the containment. The FPs initial inventory provided to ISODOP for the calculation of decay heat and of isotopes transmutation has been estimated with an ORIGEN-ARP code (Rearden and Jessee, 2018) calculation, by assuming a four-year fuel cycle lifetime, for an average burnup of 40,000 MWd/tU. The corresponding FPs inventory (and decay heat) is distributed in the core considering the axial - radial profiles using dedicated factors in the FPEVOL structure of the code.

4. ASTEC simulation of DBA and BDBA sequences

4.1. Simulation of DBA sequence

A simulation of the reference DBA sequence for the generic IRIS reactor has been performed after 2000 s of steady-state simulation, needed to reach the reactor nominal conditions. The DBA transient is initiated (t = 0 s) at the opening of the DVI break. The calculation has been carried out for 70000 s and the results have been described and discussed in previous works (Maccari et al., 2021; Maccari et al., 2021).

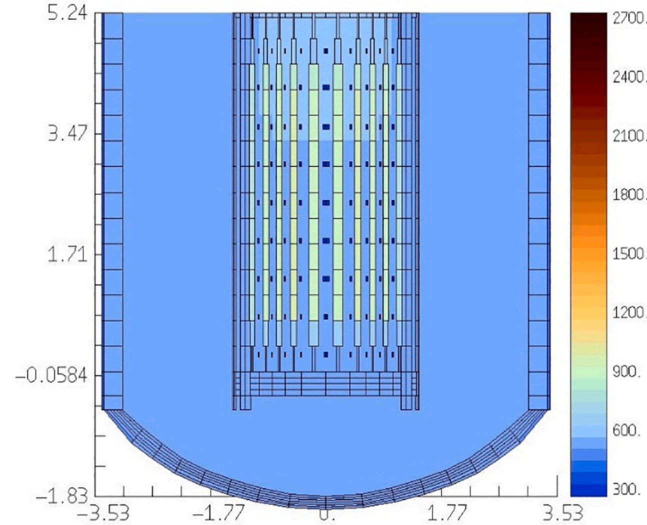

Fig. 4. Scheme of ICARE IRIS reactor nodalization of the RPV and core (Maccari et al., 2021).

The main timings of the DBA simulation carried out with ASTEC V2.2 beta have been summarized in Table 1.

4.2. Simulation of BDBA sequences

4.2.1. Assumptions for the BDBA simulations

Considering as reference the DBA simulation sequence, four BDBA scenarios have been assumed by starting from the same steady-state simulation and initiator event. In each scenarios it is assumed the failure of different selected passive safety systems, which would be activated by the opening of valves at specific set-points. Indeed, despite passive systems work thanks to natural-driven forces, their activation is based on electric signals and the opening of valves that, as an extreme hypothesis, have been considered to fail. The 4 assumed BDBA scenarios are: failure of EBTs, failure of EBTs and LGMS, failure of EHRS and ADS st-1, failure of EBTs, LGMS, EHRS and ADS. The activations of the

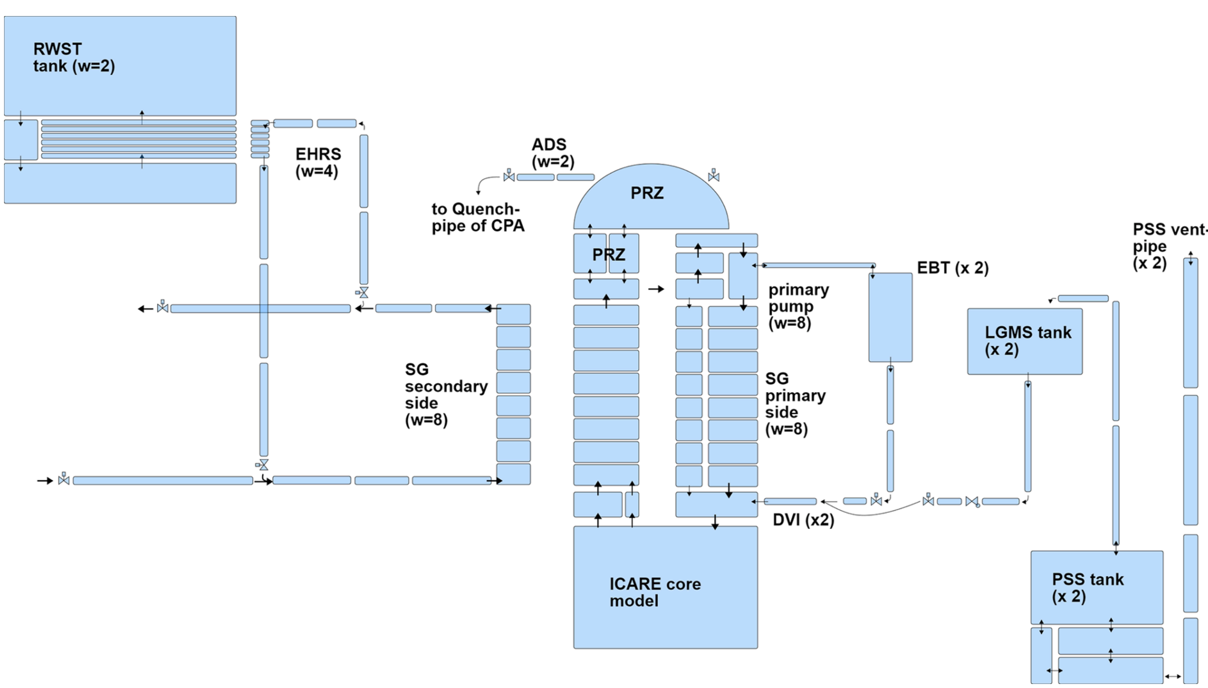

Fig. 3. Scheme of CESAR IRIS reactor nodalization of the PCS, secondary coolant system and passive safety systems (Maccari et al., 2021).

Table 1
Timings of main events in the 4 BDBA sequences and the reference DBA.

Event [s]	Reference DBA	EBT failure	EBT, LGMS failure	EHRS, ADS st-1 failure	All failure
Break	0	0	0	0	0
High Containment P. set-point	29	29	29	29	29
Low PRZ P. set-point	134	134	134	302	302
EBT-A empting	320	–	–	580	–
Low ΔP RPV-Cont. set-point	1330	1375	1375	16905	26960
EBT-B empting	1650	–	–	2400	–
RC level at DVI level	4100	5200	5200	14900	74000
Low LGMS mass set-point	17800	18000	–	132150	–
LGMS-A empting	20600	21500	–	135000	–
LGMS-B empting	25500	25600	–	155000	–
RWST water boiling	48000	48000	55000	–	–
Cladding failure	–	–	–	7800	9170
Corium relocation in LP	–	–	–	91560	111600
Containment failure	–	–	–	133950	113050

considered safety systems follow the same control logics of the reference DBA (see Section 2.1). The passive safety systems that guarantee the operability in all the scenarios are: PSS system, opening of the RI-DC valves. Furthermore, the thermal–hydraulic coupling between RPV, containment and PSS can be considered as a passive mitigation strategy that is always available. The reactor containment failure is assumed in case of internal pressure greater than 13.5 bar (design pressure of an IRIS-like containment). In this condition, a large breach is assumed to open and to directly connect the top part of the DW to the environment, determining the containment depressurization to the atmospheric pressure. The 4 calculations have been initiated by starting from the same steady-state; the opening of the 2-inch break of DVI line-A is considered to take place at the Start Of the Transient (SOT) (t = 0 s). The main timings of the simulation results are summarized in Table 1 and the plots of the main Figure of Merit (FOMs) have been reported from Fig. 5–10 against the reference DBA results.

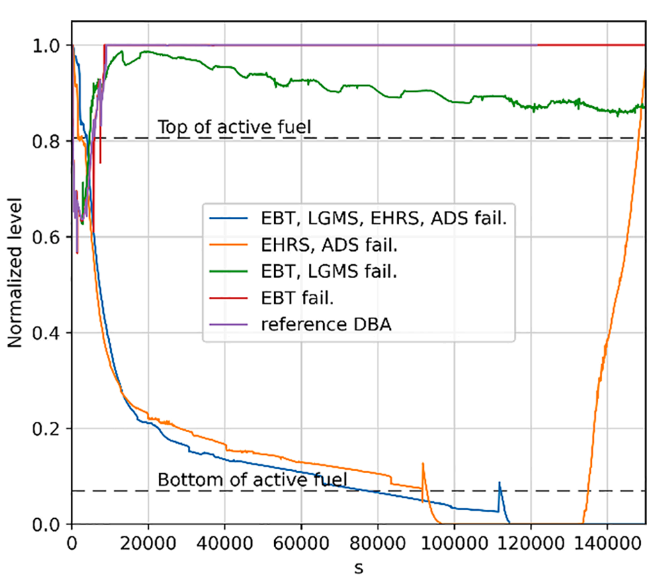

Fig. 5. Normalized value of averaged collapsed level in the core (upper and lower core plates elevations taken as reference).

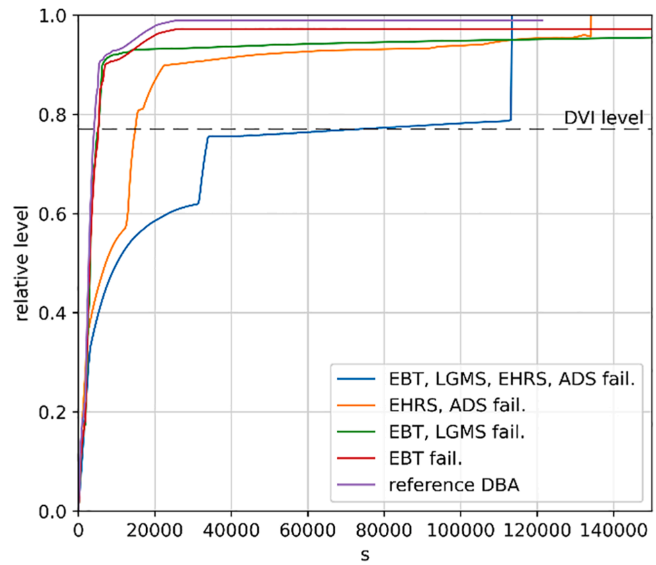

Fig. 6. Normalized value of water level in the RC, with respect to the RC total elevation.

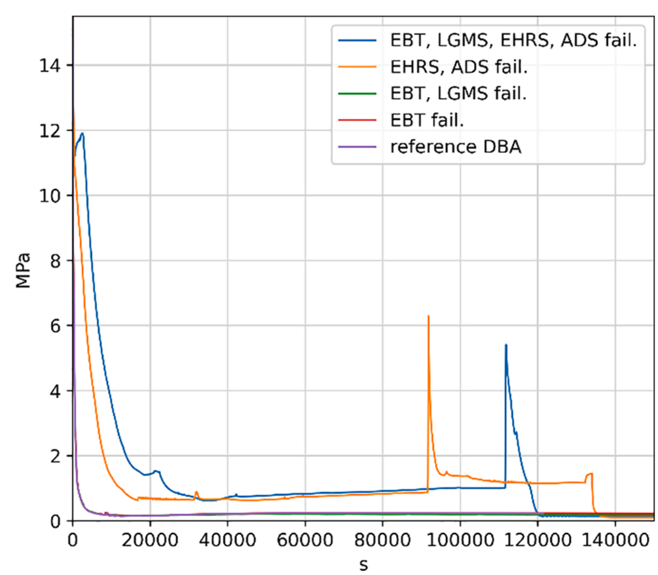

Fig. 7. PCS pressure (PRZ).

4.2.2. Results of EBTs failure scenario

From a qualitative and phenomenological point of view, the EBTs injection failure scenario is similar with respect to the reference DBA sequence. The lack of EBT cold water injection determines a slight reactor cooling and depressurization delay as shown in Table 1. A slightly lower RC water level at the end of the transient is observed due to the lack of EBTs injection.

4.2.3. Results of EBTs and LGMS failure scenario

The present transient shows a qualitative trend of the core water level decrease during the RPV depressurization and core refill (Fig. 7) preventing any core heat-up (Fig. 9). The RPV depressurization effect of the ADS st-1 openings and the EHRS power removal mitigates the coolant blowdown from the break. The EHRS in natural circulation permits the core cooling preventing any core damages. As shown in Fig. 5, due to the lack of LGMS injection, the final core level is lower than

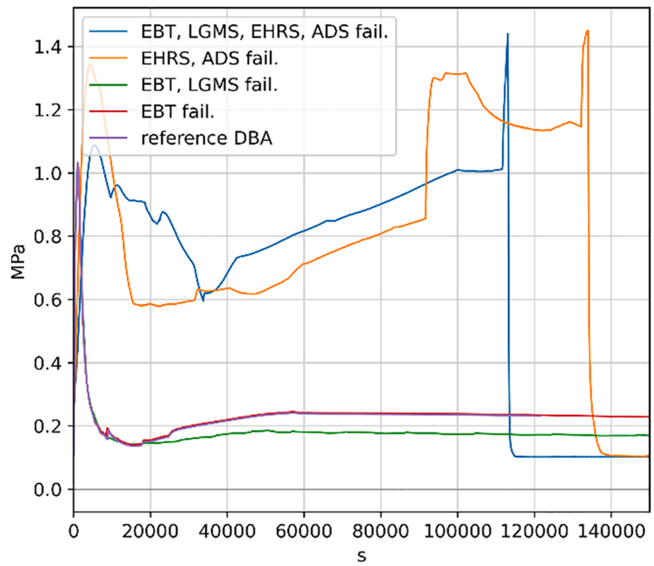

Fig. 8. Containment pressure.

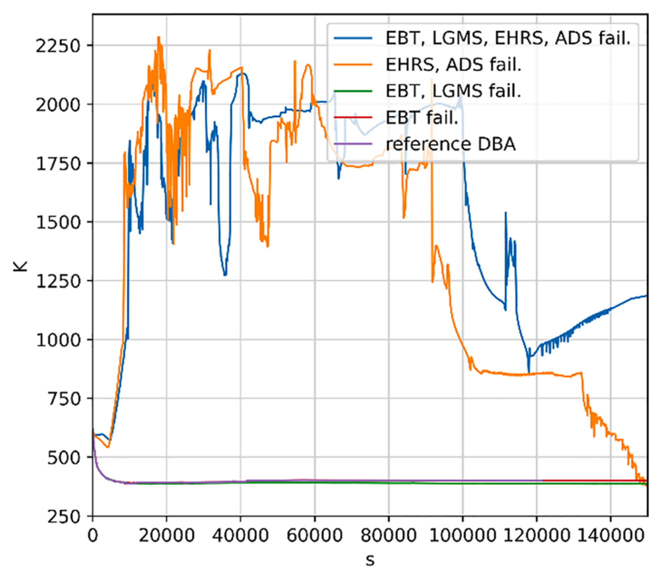

Fig. 9. Core outlet temperature.

the previous scenarios (DBA and EBTs failure). At the end of the simulated transient, the water level is above the top of active fuel, slowly keeping to decrease due to the coolant evaporation. With respect to the previous scenario, a phenomenological difference is related to the coupling between RPV and containment: considering that the water level in the RC reaches and overcomes the DVI level with very similar timing (Fig. 6, Table 1), it has to be observed that the present transient does not feature a backward water flow through the break keeping the RPV water level at the same level as the RC (Fig. 5 and Fig. 6). It should be attributed to the lack of opening of ADS st-2, activated by “LGMS low water mass signal” never reached in this scenario (Table 1). Indeed, in the present scenario only ADS-st1, connecting the PRZ to the pool of the QT, valves have been opened. Considering the smaller size of ADS st-1 compared to the ADS st-2, a small over pressurization of RPV with respect to containment (of about 0.3 bar = 3 m of water head) is observed. For this reason, no backward water flowing through the break is simulated by the code in the present calculation.

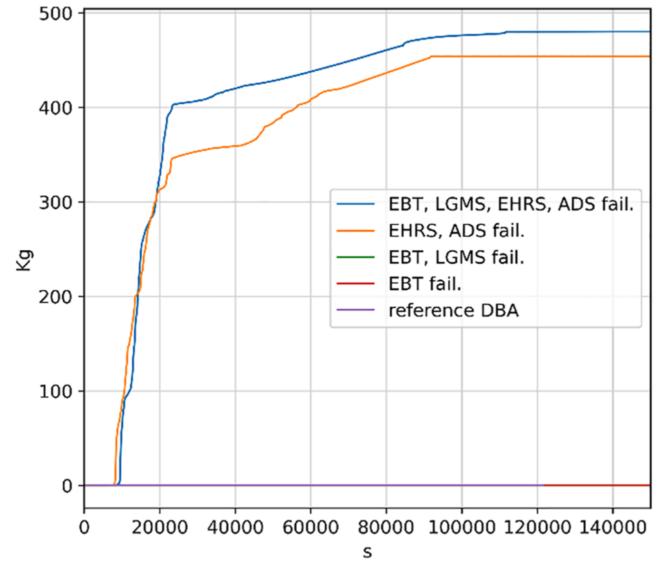

Fig. 10. Total hydrogen mass production.

4.2.4. Results of EHRS and ADS st-1 failure scenario

The lack of ADS-st1 opening and of EHRS heat removal determines the delay of the containment-RPV low differential pressure set-point ($\Delta P < 0.5$ bar) activation, as shown in Table 1. As a consequence, the water leak through the break does not decrease soon enough to avoid the core uncovery in the first part of transient (0–10000 s). The impossibility to cool the uncovered core (despite the intact EBT injection) leads to the start of degradation at around 7800 s after the SOT. The RPV–containment low DP-signal arrives at about 16905 s after the SOT (Table 1), opening at this point the LGMS valves. During the core degradation, steam and incondensable gases produced and the lack of the EHRS cooling maintain the containment and the RPV pressure between 6 and 9 bar, as can be observed in Fig. 7 and Fig. 8. The higher RPV pressure (with respect to the safety systems line) prevents any injection of LGMS water into RPV (through the intact DVI line); while a slow LGMS injection into the containment (through the broken DVI line) takes place. In addition, the RC water level, which has reached the break elevation at 14900 s after the SOT, is not allowed to refill the core due to the RPV pressurization. Core degradation advances until the bottom support plate failure and corium relocation in the LP (slumping) takes place at 128100 s after the SOT. The vaporization of LP water causes a very high-pressure peak in the RPV (about 63 bar), which is followed by a steep containment pressurization, up to around 13 bar, very close to the containment design pressure of 13.5 bar. After the slumping, the systems feature a stable phase with RPV pressure at around 13 bar, during which the corium is retained in the LP externally cooled by the RC water. At 111600 s after the SOT, the low water mass in the LGMS tank-A triggers the “Low LGMS mass set-point”, opening the ADS st-2 valves. As a consequence, the RC water is allowed to flow back through the break, interacting with the hot corium pool and, hence, producing overheated steam which quickly increases again the RPV and the containment pressures. As a consequence, the containment fails at 113050 s and this leads to a fast depressurization of all the reactor systems to the atmospheric pressure and, as expected, by a large release of hydrogen, steam and FPs to the environment zone. The ASTEC visualization of the core degradation is reported in Fig. 11, considering four timings of the core degradation sequence. The RC water keeps entering in the RPV through the break, completely filling the core. The final state of the reactor, at 150000 s after the SOT, features the corium retained in the LH and submerge by water, as reported in the last time of Fig. 11. The possibility of an In-Vessel Melt Retention (IVMR) strategy within this final configuration (internal and external core cooling) should be investigated in further analyses. From a preliminary study, there seems to be

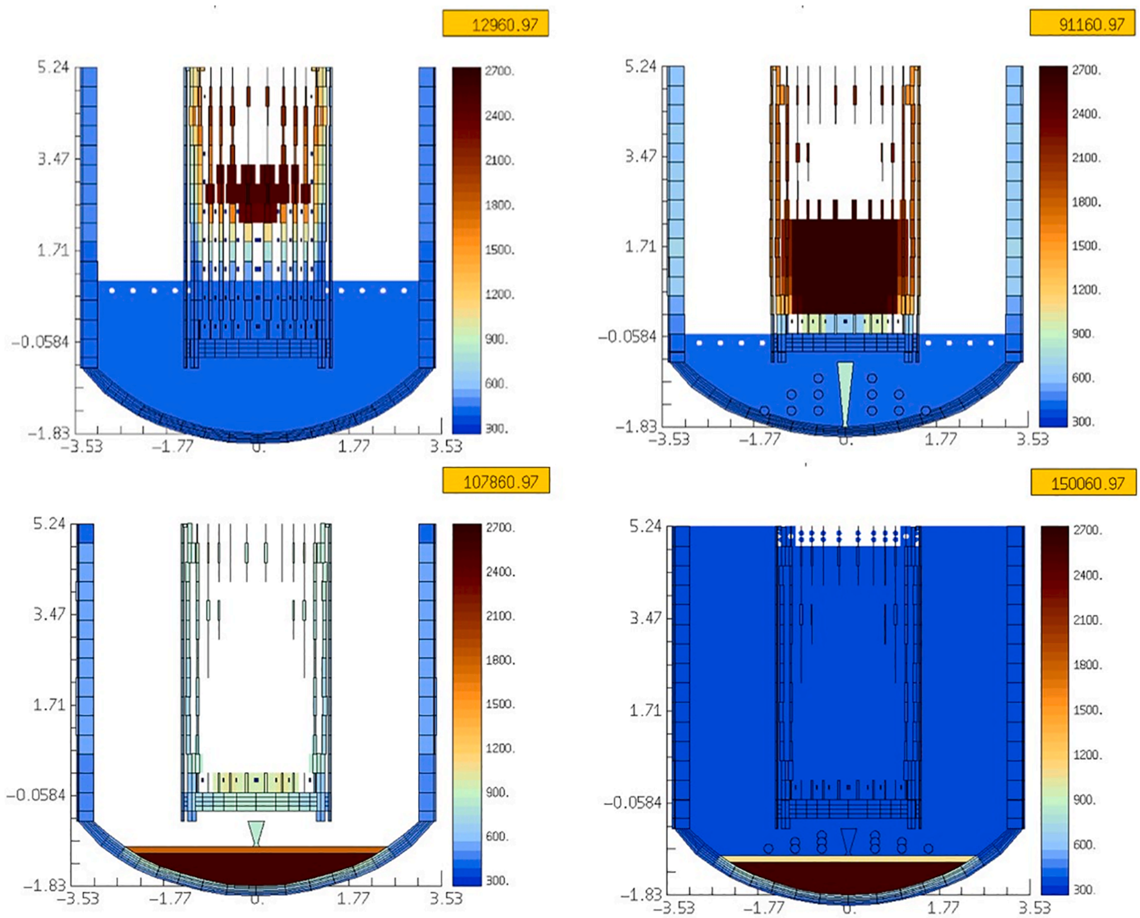

Fig. 11. ASTEC - ICARE mask of core degradation evolution for the ADS st-1 and EHRS failure scenario.

promising possibilities to retain the corium in the LP by both internal and external cooling.

4.2.5. Results of EBTs, LGMS, EHRS and ADS st-1 failure scenario

As can be inferred at first sight of Fig. 5 and Fig. 10, the present scenario presents a similar phenomenological behavior to the one above described (failure of EHRS and ADS stage-1). The first phases of transient, before the onset of core oxidation, features some discrepancies with respect to the previous scenario: due to the lack of the EBTs cold water injection in RPV, the core temperature and primary pressure remain slightly higher; while, on the contrary, the containment pressure increase is much stronger in the previous scenario, where the pressurized water of the EBT (at the primary pressure) connected to the broken DVI is flashed in the containment. The onset of core oxidation and degradation arrives within a similar timing in the two SA scenarios, as can be inferred by looking at Table 1 and at the hydrogen generation in Fig. 10. The core degradation evolution of the present scenario features a very similar qualitative behavior to the previous one, as can be observed by looking at the hydrogen production (oxidation processes) and at the temperature evolution, in Fig. 9 and Fig. 10 respectively. The slumping arrives 20000 s later in this last scenario with respect to the previous one and a similar primary pressure peak is predicted by the code in the RPV, as shown in Fig. 7. The containment pressure increase follows the fast RPV pressurization. However, in this case, due to the initial higher pressure before the corium slumping, the containment reaches the failure pressure of 13.5 bar at this point, determining an earlier opening to the environment and depressurization. After the containment failure, the lack ADS-st2 valves opening makes the RPV pressure to remain higher than the containment pressure along all the sequence (with a minimum over-pressurization value of 0.4 bar). As a consequence, a reverse flow through the break refilling the core is never predicted in this scenario. At the end of the sequence (150000 s), the corium is retained in the LP and the LH structures is cooled only by the external water in the RC (Fig. 12). It is evident that margins are reduced in this configuration compared to the previous case with both in and ex-vessel cooling.

5. Conclusions

The target of the paper is to provide the first insights about the capability of the ASTEC model of the generic IRIS reactor to be used in BDBA and in SA analyses, if significant core degradation takes place. The scenarios selected is the double ended break of one DVI line and it has been simulated with the ASTEC V2.2 beta version. In order to determine BDBA and SA scenarios the not operation of selected passive system have been postulated. In the present study, four BDBA simulations have been carried out and the actuation failure of different selected passive systems has been postulated as following: failure of EBTs; failure of EBTs and LGMS; failure of EHRS and ADS st-1; failure of EBTs, LGMS, EHRS and ADS. From the analysis of the results, the accident mitigation achieved in the first scenario (EBTs failure) and the partial mitigation of the second scenario (failure of EBTs and LGMS) show a safety margin guaranteed by the passive mitigation strategy even in the case of multiple failures of the passive injection systems. Accordingly, the results underline the crucial role played by EHRS and ADS systems in the passive mitigation strategy: the actuation of ADS st-1 ad EHRS guarantees the depressurization of the PCS and the pressurization of containment (in safety conditions), anticipating the pressure equalization between the two systems and mitigating the blowdown of coolant from RPV and, consequently, avoiding core damages. The following decrease of

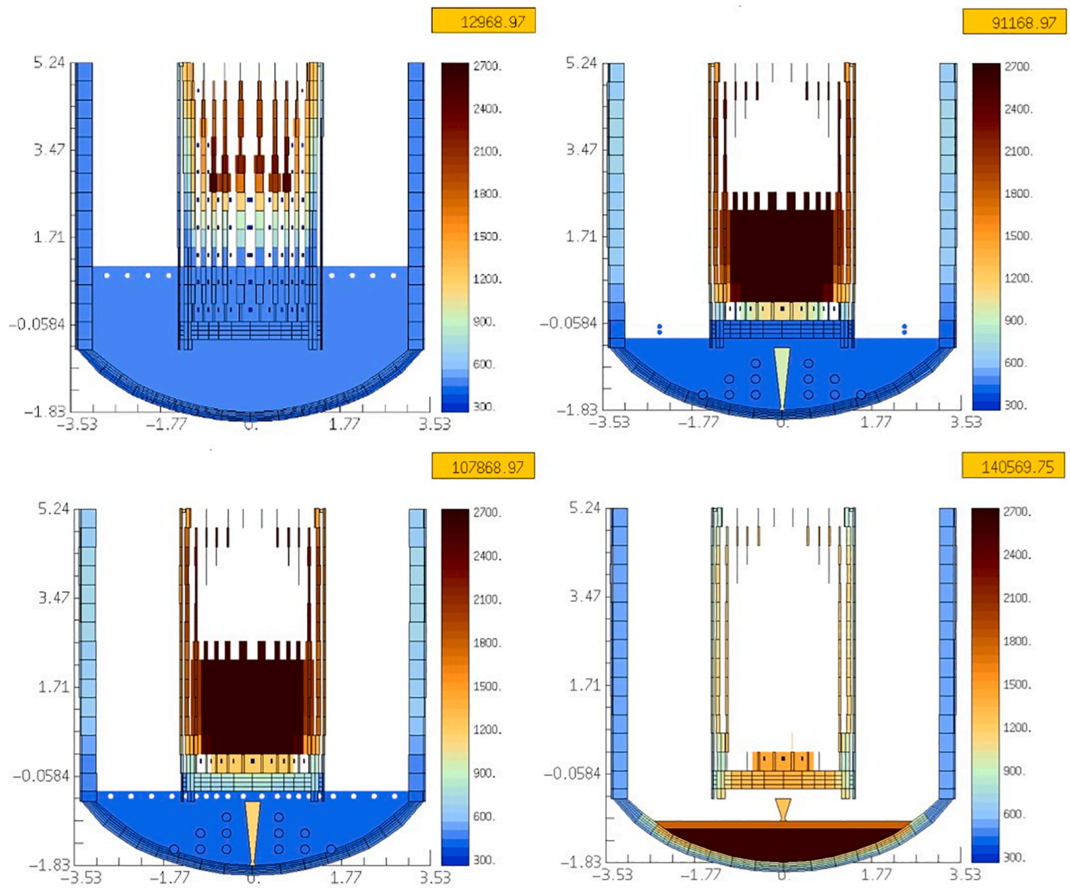

Fig. 12. ASTEC - ICARE mask of core degradation evolution for the EBTs, LGMS, EHRS and ADS st-1 failure scenario.

containment pressure below the PSS pressure guarantees the RC flooding (from PSS vent-pipes) and the LGMS injection. The effect of ADS st-2 is also highlighted: in the analyses, the core refill from the break is predicted to take place only in case of ADS st-2 opening (EBTs failure scenario; EHRS, ADS st-1 failure scenario). If ADS st-2 valves are closed, indeed, the RPV remains pressurized and water cannot flow back through the break from the RC.

Two IVMR configurations have been observed in the described SA scenarios (Figs. 11 and 12). The first one, obtained in the case-study with EBTs, LGMS, EHRS and ADS st-1 failure, is a traditional IVMR configuration in which most of decay-heat is removed by the external RC water thorough the LH structure, and a corium stratification with focusing effect is present. The second IVMR configuration in this study is obtained when the EHRS and ADS st-1 fail, and it features a corium cooling from both the external water in RC and the internal water on the top of the corium pool, removing heat directly from the top metallic layer. The large heat removed by water from the top of the corium pool could mitigate the focusing effect and may prevent from high heat-flux to a limited LH surface. Further investigations are necessary to characterize the ASTEC code IVMR modelling capability in integral SMR configuration, e.g. the code capability to predict heat transfer between the melt and the top water during a IVMR with top water cooling.

More in general, the present study is important to highlight as a postulated SA sequences in integral passive SMR designs may by characterized by specific phenomenology (e.g., IVMR with top water cooling, systems coupling during core degradation, etc.) not observed before for larger size reactors. Such phenomena, due to the peculiarity of SMRs (integral geometry, tight thermal–hydraulic coupling between primary system and containment, operation of passive systems in BDBA sequences, etc.), influence the evolution of the SA sequence and, therefore, needs to be accurately predicted by SA codes. In addition, the failure of the containment in two of the reported scenarios underlines the importance to further study SA transient in smaller containment (characterized by a reduced containment-volume/core-power ratio compared to large-LWR), including hydrogen risk.

About the ASTEC performances, the activity gives the first insights about the capability of the code to predict different accidental sequences due to the postulated not-operation of the selected passive systems in a generic SMR. Validation activity is necessary in order to quantitatively assess the code capability for simulating SMR configurations and related phenomenologies.

CRediT authorship contribution statement

Pietro Maccari: Conceptualization, Software, Formal analysis, Investigation, Data curation, Writing – original draft, Writing – review & editing, Visualization, Methodology. **Giuseppe Agnello:** Data curation, Writing – original draft, Writing – review & editing, Visualization. **Fulvio Mascari:** Conceptualization, Methodology, Investigation, Resources, Writing – review & editing, Validation, Supervision, Project administration, Resources. **Stefano Ederli:** Software, Methodology, Writing – review & editing, Validation.

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.

Data availability

The data that has been used is confidential.

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