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ANALYSIS OF ACCIDENT SCENARIOS IN A GENERIC NATURAL CIRCULATION SMALL MODULAR REACTOR WITH THE ASTEC CODE IN THE FRAMEWORK OF THE SASPAM-SA PROJECT

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Abstract. In response to the needs for safe and sustainable energy solutions, the emergence of Small Modular Reactors (SMRs) has increased significant interest. The HORIZON-EURATOM SASPAM-SA project, currently ongoing, addresses the transferability of knowledge from large Light Water Reactors (LWRs) to integral Pressurized Water Reactors (iPWRs) and analyses the behavior of generic SMR designs with passive mitigation strategies under accident conditions. In this framework, this study investigates the safety assessment of a generic natural circulation iPWR using the code ASTEC (Accident Source Term Evaluation Code) developed by the French Institute for Radiation Protection and Nuclear Safety (IRSN). The ASTEC code is able to simulate an entire accident sequence in a nuclear power plant, from the initial event to the potential release of radioactive elements outside the containment, involving the modelling of diverse thermal hydraulic and physico-chemical phenomena. This paper presents the outcomes derived from the analysis of hypothetical Design-Basis Accident (DBA) and Severe Accident (SA) scenarios, in one of the two generic iPWRs considered in the SASPAM-SA project, showing the code's capability to adequately describe the key thermal hydraulic and core degradation phenomena.

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1 INTRODUCTION

Safe, sustainable and reliable energy solutions have become increasingly urgent to face climate change and global energetic needs. Nuclear power, once sidelined due to the past incidents, is gaining a new resurgence as alternative to fossil fuels with new fuel materials technology, as well as new design concepts. Small Modular Reactors (SMRs) are the key elements in this new paradigm addressing many concerns related to conventional large-scale nuclear power plants (NPPs).

SMRs are designed to be more flexible and safer than traditional reactors. These new reactors are typically designed to generate one-third to one-fifth of the power output of conventional NPPs, making them easier to cool down in case of an incident due to the lower decay heat produced. Their smaller size also allows for the integration of advanced safety feature, such as passive cooling systems that can operate without external power or human intervention, which is a significant advantage in the event of a power outage or natural disaster. These features make SMRs particularly suitable for deployment in remote areas, as well as for providing reliable backup power to support the intermittent nature of renewable energy sources like wind and solar (Cho and Lee, 2024). According to the Nuclear Energy Agency (NEA) (2023), there are more than 50 SMR projects underway worldwide, demonstrating the broad interest in this technology.

In this context, the EU-funded SASPAM-SA project aims to explore the applicability and transferability of operational expertise from large light-water reactors to Integral Pressurized Water Reactors (iPWRs), with focus on addressing the European licensing needs for the SAs and Emergency Planning Zones (EPZs) (European Commission, 2022; SASPAM-SA, 2024). Part of the SASPAM-SA project is the assessment of the safety of a generic natural circulation iPWR using the Accident Source Term Evaluation Code (ASTEC) developed by the French Institute for Radiation Protection and Nuclear Safety (IRSN).

The code ASTEC is able to simulate a complete accident sequences in nuclear power plants, starting from the initial event to the potential release of radioactive materials. It captures a broad spectrum of thermohydraulic and physico-chemical phenomena, enabling a thorough safety assessment of nuclear reactors.

In this work, the analysis of hypothetical Design-Basis Accident (DBA) and Severe Accident (SA) scenarios in a generic iPWR using ASTEC is presented.

2 IPWR DESIGN AND MODEL NODALIZATION

The generic iPWR design analyzed in this work is a 160 MWth natural-circulation pressurized water reactor. It features a shrouded reactor core, a riser, a pressurizer, and two Helical-Tube Steam Generators (HTSGs) within an integral Reactor Pressure Vessel (RPV), all enclosed by a steel containment vessel submerged in a large pool of borated water, serving as the ultimate heat sink. The high coolant inventory in the RPV, combined with the immersed containment in the pool, provides substantial thermal inertia, while the vacuum atmosphere inside the containment enhances steam condensation in case of steam release from the RPV. The reactor core includes 37 fuel assemblies in a 17x17 array with an active height of approximately 2 [m], and the riser is surrounded by a downcomer housing the HTSGs (SASPAM-SA, 2023).

The iPWR analized includes two main safety systems:

- Emergency Core Cooling System (ECCS): This system manages heat removal via three reactor venting valves (RVVs) at the top of the RPV. Steam condenses on the containment vessel surface, and the water flows back to the reactor core through two reactor recirculation valves (RRVs), ensuring core cooling even in the event of multiple valves failure.

- Containment Heat Removal System (CHRS): Transfer energy to the pool by conduction and convection modes.

ASTEC simulates the entire sequence of a SA in a nuclear power plant, specifically in a water-cooled reactor. It covers from the initial event to the release of radioactive elements outside the containment, involving the modelling of diverse physico-chemical phenomena (Chatelard et al., 2016). At the Karlsruhe Institute of Technology (KIT), the version 3.1 of ASTEC was used in order to assess the simulations. The code comprises of several integrated modules in charge of simulating the main phenomena that takes place during the defined sequences. These modules include: ICARE describes the in-vessel degradation phenomena up to the vessel bottom head failure, CESAR simulates the two-phase thermal-hydraulic in the Reactor Coolant Systems (RCS), CPA computes the thermal-hydraulic within the containment, ISODOP calculates the isotopes masses, the decay heat of fission products (FP) and activity of elements in different zones of the reactor and in the containment, SOPHAEROS is dedicated to the FP and structural materials (SM) transport phenomena, among others.

The original input deck with the reactor model and conditions was provided by TRACTEBEL for version 2.2 and then upgraded to version 3.1 by KIT in collaboration with IRSN. The core is discretized in 2D, defining 22 levels axially and 6 radial channels. Of the axial levels, the first level across all channels correspond to the lower plenum mesh, 17 levels contains fuel rod material (fuel and cladding) and the rest correspond to structural material elements. Radially, the four inner channels contain 1, 8, 12 and 16 fuel assemblies, respectively. The fuel inventory was delivered during the project based on End-of-cycle conditions (EOC). The fifth and sixth channels corresponds to the Bypass and Downcomer, respectively. The pool domain was divided into the nuclear power module (NPM) Pool, which corresponds to the volume surrounding the containment, while the Reactor Building includes the rest of the pool. The containment was divided in three main volumes.

The schematic nodalization used to discretized the domain of the iPWR is presented in the Fig. 1.

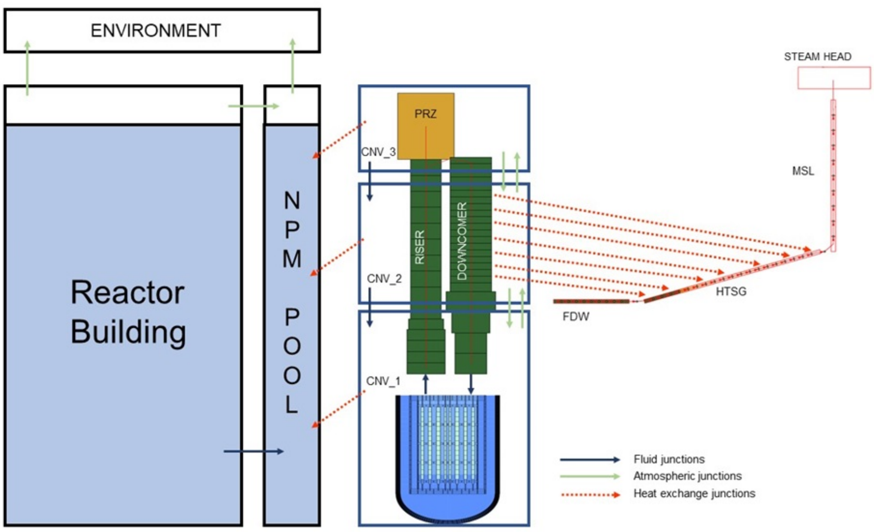

Figure 1: Schematic nodalization of the iPWR Design model in ASTEC V3.1

3 HYPOTHETICAL SCENARIOS DESCRIPTION

From the wide conditions analysed, a DBA and a SA events are presented in this work.

The postulated DBA event consisted on a break on the Chemical and Volume Control System (CVCS) discharge line at time $t_0 = 0$ [s], connecting the Downcomer with the containment. The break is located at 3.5 [m] from top of active fuel (TAF). The reference break size was set to 4.3 [cm] in diameter. In addition to this break, three more were included in this study given by 35%, 20% and 10% of the reference area.

The DBA scenario is comprised of:

- Loss of AC power (Feedwater, CVCS);
- Main Steam Line (MSL) closure ($t_0 + 5s$);
- ECCS activation with a pressure difference RPV-Cont. < 6.2 [MPa] and Cont. water level > 5.8 [m];
- 3 RVVs and 2 RRVs available.

The SA event used has the same conditions as the DBA scenario with break size of a diameter of 4.3 [cm], except for:

- The break is located 5.3 [m] from TAF to avoid the reverse flow;
- ECCS activation with a pressure difference RPV-Cont. < 6.9 [MPa] and Cont. water level > 5.8 [m];
- 2 RRVs not available and stuck closed.

Parameter	value
Core Power [MW]	160.00
Feedwater Temperature [K]	422.0
Feedwater Pressure [MPa]	3.8
Average Core Temperature [K]	560.0
Burnup [MWd/kgU]	60.0
Containment Temperature [K]	320.0
Pool Temperature [K]	320.0
Pool water level [m]	16.8

Table 1: Initial conditions for the steady-state simulation (SASPAM-SA, 2023)

A steady state calculation was carried out using the code ASTEC v3.1 and the fission products inventory in a EOC condition with a burnup of 60 [MWd/tU]. The initial conditions to set the start of the calculation are listed in Table 1 and the code reached the steady-state condition successfully.

4 RESULTS

4.1 DBA Analysis

In this section, the main results for the DBA scenarios analysed are presented with a comparison between the behaviour for the different break sizes. The simulations were run up to 15000 [s] from the initiating event.

Parameter	ASTEC			
Relative break area size [%]	100	35	20	10
SCRAM time [s]	4	7	13	30
ECCS Actuation [s]	340	1285	2935	7695
RVVs and RRVs opening time [s]	345	1290	2940	7700
Max. liq. level in the containment [m]	6.6	6.8	6.9	7.3
Max. pressure in the containment [MPa]	4.5	3.9	3.5	3.0

Table 2: Summary of event sequences for the DBA scenarios calculated with ASTEC.

Table 2 presents the key parameters obtained from the simulations. First and foremost, it is observed that the SCRAM time increases as the break size decreases, owing to the slower pressure rise within the containment for smaller break sizes. In the cases of 100%, 35%, and 20% break sizes, the SCRAM signal is initiated because the containment pressure exceeds 62 [kPa]. On the contrary, in the case of a 10% break, the SCRAM signal results from a minor initial pressurization within the Pressurizer, exceeding 13.8 [MPa], although this pressure rapidly decreases thereafter. Similarly, as the break size decreases, the ECCS activation time increases, as well as the maximum water level in the containment. On the contrary, the peak pressure reached in the containment decreases with the decreased in the break size.

The evolution of the mass flow rate through the break, for both water and steam, is shown in Fig. 2 A) and B), respectively. As it can be seen, the maximum water flow rate occurs when the break take place and its value decreases as the accident progresses, the water level in the downcomer and the pressure in the RPV decrease. As it is expected, the larger the break, the higher the mass flow through the break and it drops when the water level in the RPV falls below the break elevation. For the steam flow, it starts when the void fraction increases in the system and drops suddenly when the ECCS is activated and the RVVs are opened. Similarly, the maximum steam flow achieved decreased as the break decreases.

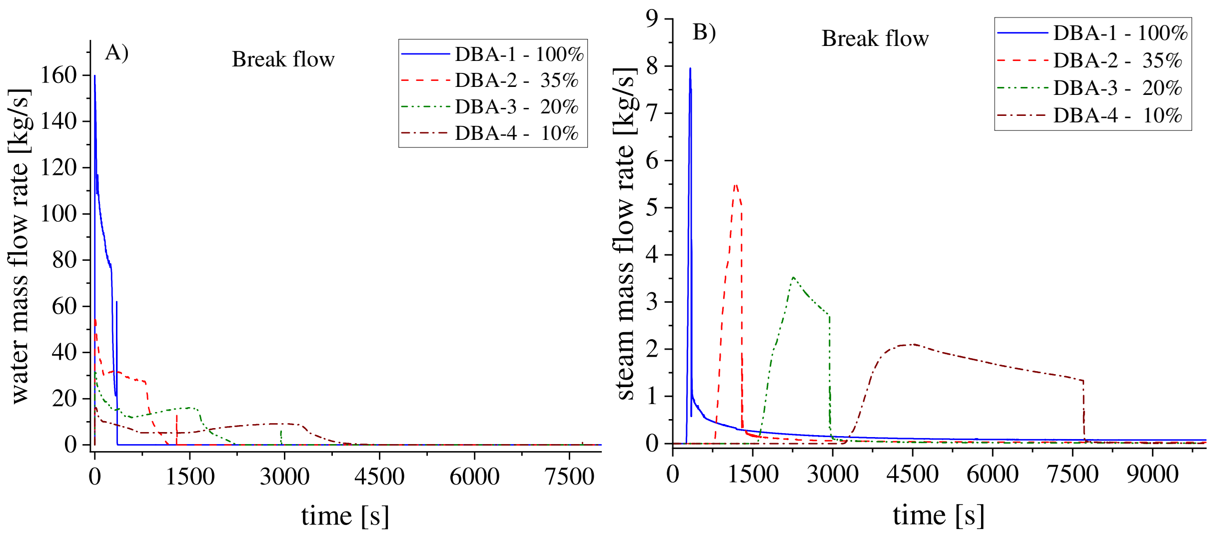

Figure 2: A) Water mass flow rate and B) steam mass flow rate evolution through the break of different sizes in the DBA case proposed.

The water level evolution in the containment for the four different breaks of the DBA scenario is presented in Fig. 3 A). The rate of water accumulation in the containment is slower for smaller break sizes due to the lower water flow rate, as shown in Fig. 2 A). However, the maximum level, which is reached just before the RVVs open, occurs at smaller break sizes, being 7.3 [m] for the 10% break. Fig. 3 B) displays the evolution of the water level in the RPV above the TAF (Top of Core in reality) for the simulated break sizes. As it is shown, the water level remains above the TAF throughout the scenario at 2 [m], approximately, thus meeting the Design-Basis Accident (DBA) concept.

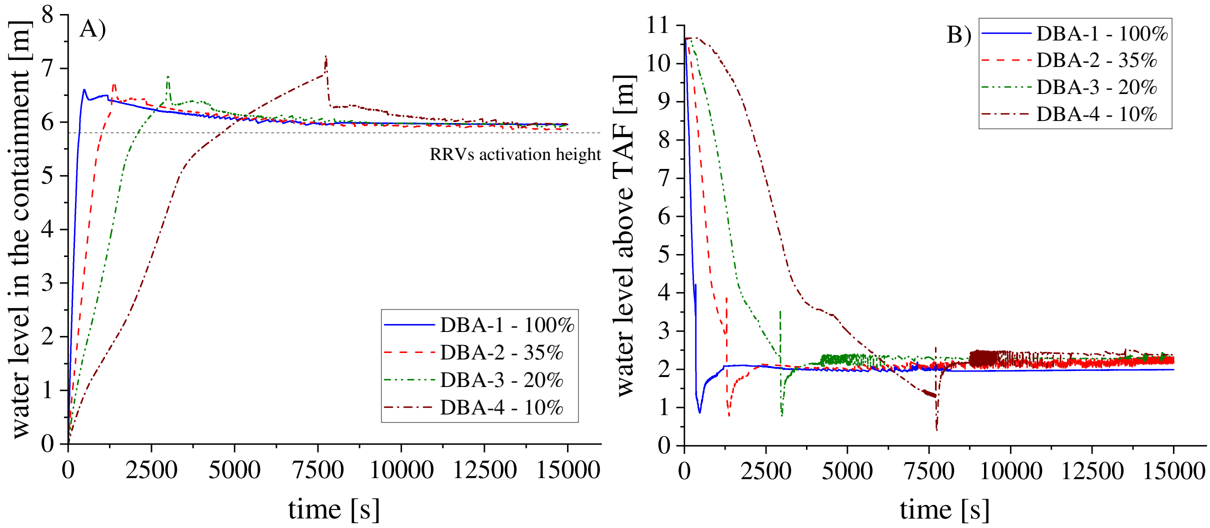

Figure 3: Evolution in time of water level A) in the containment and B) in RPV above TAF for different break sizes in the DBA case proposed.

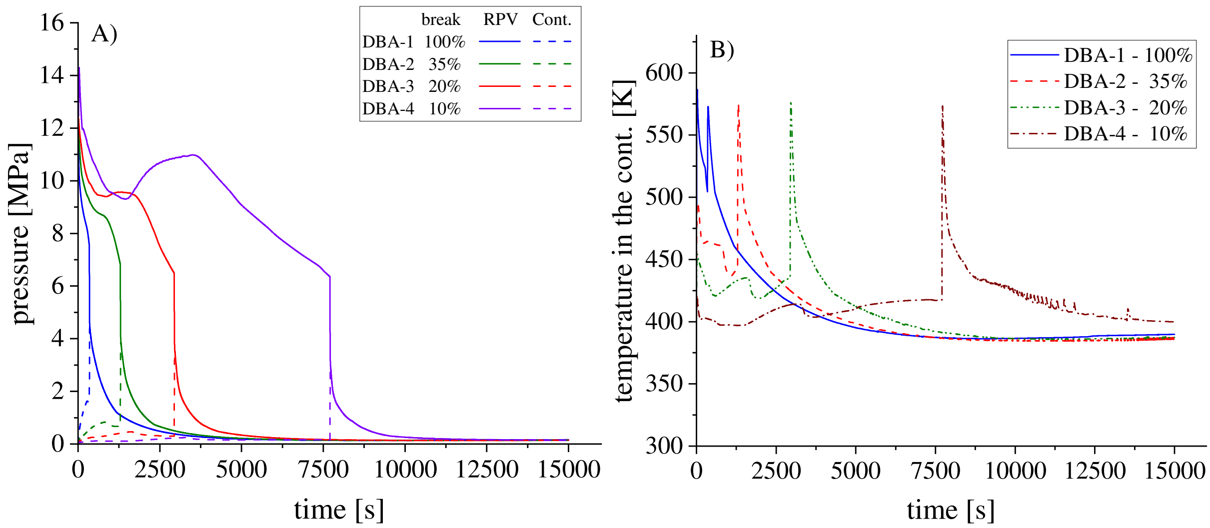

Figure 4: A) RPV and containment vessel pressure evolution and B) containment temperature evolution for different break sizes in the DBA case proposed.

In Fig. 4 A), the evolution of pressure in both the containment vessel and the reactor pressure vessel is presented. Here, it can be observed that for all cases, except the 100% break, there is an initial drop in the RPV pressure followed by a repressurization phase. This is followed by a continuous drop in pressure until the activation of the ECCS, which includes the opening of the RVVs and RRVs, leading to the equalization of pressure with that of the containment. The temperature evolution in the containment vessel for all the DBA scenarios analyzed are presented in Fig. 4 B). The temperature shown corresponds to the gas mixing temperature in the upper zone of the containment nodalization. Two significant temperature peaks are observed: the first one is presented when the break takes place and the second one occurs during the activation of the ECCS and the opening of the RVVs. During the first peak, the temperature reaches a maximum value of 585 [K] for the 100% break, while for the second peak, similar temperature values around 570 [K] are observed for all break scenarios. After the second peak, the temperature continues to decrease below 400 [K].

It is important to note that during these scenarios, the thermohydraulic conditions are such that there is no core degradation, so no hydrogen or corium material is produced.

4.2 SA Analysis

The most relevant results of the progression of a SA with a 4.3 [cm] diameter break connecting the downcomer to the containment vessel in which the RRVs are unavailable are presented below. The simulation was run up to 100000 [s] from the initiating event.

Table 3 presents an overall summary of the results obtained with the code ASTEC showing the most relevant phenomenological events. As it can be seen, it takes around 6 hour to fully uncover the core under the conditions described and about 84% of the total hydrogen generated in the vessel is located in the steel containment at the end of the simulation.

Parameter	ASTEC
SCRAM time [s]	4
ECCS actuation time [s]	935
Max. liq. level in the containment [m]	9.8
Max. pressure in the containment [MPa]	2.7
Start/End of core uncovering [s]	5705 / 21265
H_2 onset time [s]	7535
First cladding rupture [s]	9100
Start of FPs release from fuel pellets [s]	9100
First corium slump into the lower plenum [s]	25190
Total H_2 produced in the vessel [kg]	64
Final H_2 mass in the containment [kg]	54
Final aerosol mass in the containment [kg]	323

Table 3: Summary of the relevant phenomenological events predicted by the code

In Fig. 5 A), B), C, and D), the evolution of the main thermohydraulic parameters of the system can be observed. Similar to what was previously described for the DBA cases, the maximum flow rate is established at the opening of event. The fluid starts as liquid water until the level in the downcomer decreases and the steam is present. The maximum steam flow rate through the break is approximately 10% of the maximum water flow rate. The total mass flow rate decreases considerably once the ECCS is activated and the RVVs open at ~ 935 [s]. At this time, the pressure between the containment and the vessel equalizes and continue to decrease as the scenario progresses (C). Due to the unavailability of the RRVs, no recirculation path can be established between the containment and the vessel to ensure adequate cooling of the core,

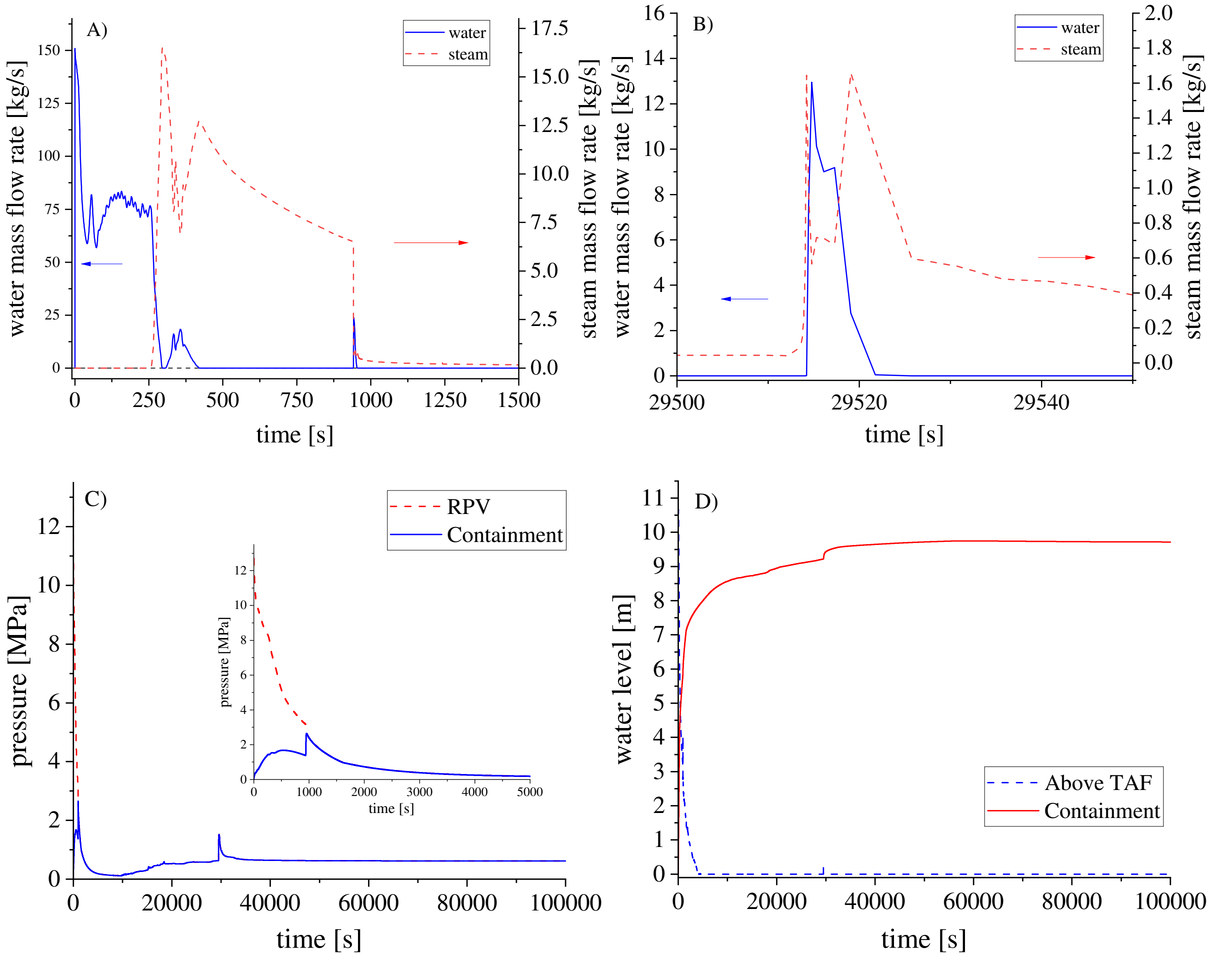

Figure 5: Evolution of the mass flow rate through the break A) (B-zoomed in relocation period), the pressure in the containment and the reactor vessel C) and water level in containment and in the vessel D) for the SA scenario analyzed.

so it starts to degrade. Then, at approximately 29500 seconds, the melted mass of core (corium) relocates into the lower plenum, generating a temporary pressure increase that can be seen in C), an increase in the fluid flow through the break in B), and consequently, an increase in the water level in the containment vessel (D). After the relocation, the system pressure decreases slowly and remains practically constant at 0.6 [MPa]. In Fig. 5 D), the water levels in the containment and in the vessel above TAF are observed. In particular, at $\sim$ 5700 [s] the core starts to uncover. while water keeps accumulating in the steel containment throughout the accident progression.

Fig. 6 A) shows the mass of hydrogen produced in the vessel by different reactions. It is observed that the main contribution is due to the oxidation of zirconium and magma in the corium. Stainless steel oxidation accounts for only 2.3% of the total production for this case, leading to an overall mass of 64 [kg]. The evolution of the selected fission products within the containment is displayed in Fig. 6 B). Here, it can be observed that the maximum amount of Xe in the containment reached 95% of the initial inventory, while I and Cs in suspended aerosols accounts for 8% and 6.5%, respectively.

Finally, in Fig. 7, the progression of the scenario is observed until 100000 [s], revealing different stages of the core uncovery and degradation. The largest relocation into the lower plenum is of $\sim$5.8 [ton]. At the end of the scenario, the corium consists mainly of a lower layer

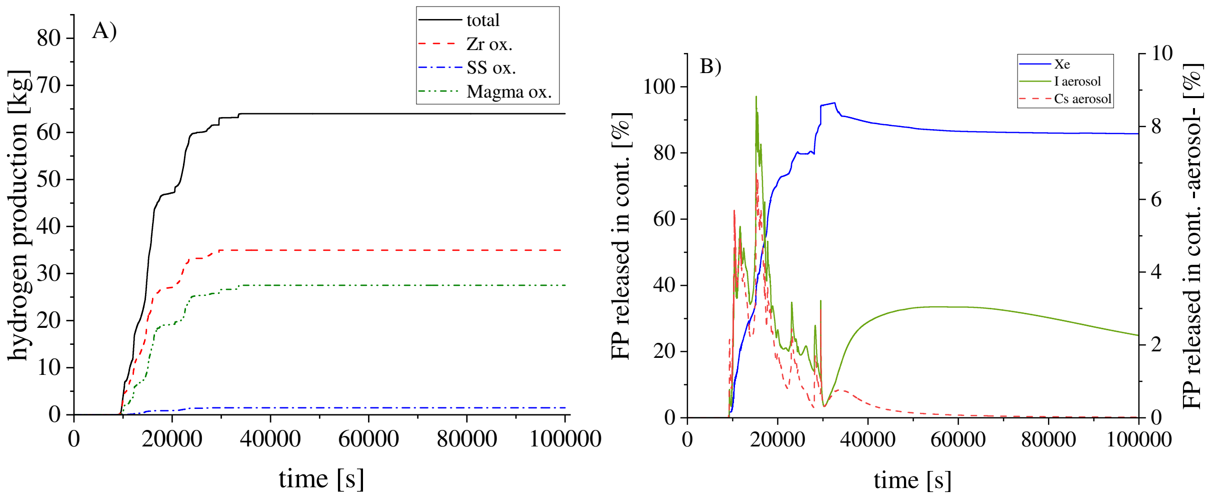

Figure 6: A) Cumulated hydrogen mass production evolution in time and its different contributions and B) FP evolution in the containment as fraction from initial inventory for the scenario presented.

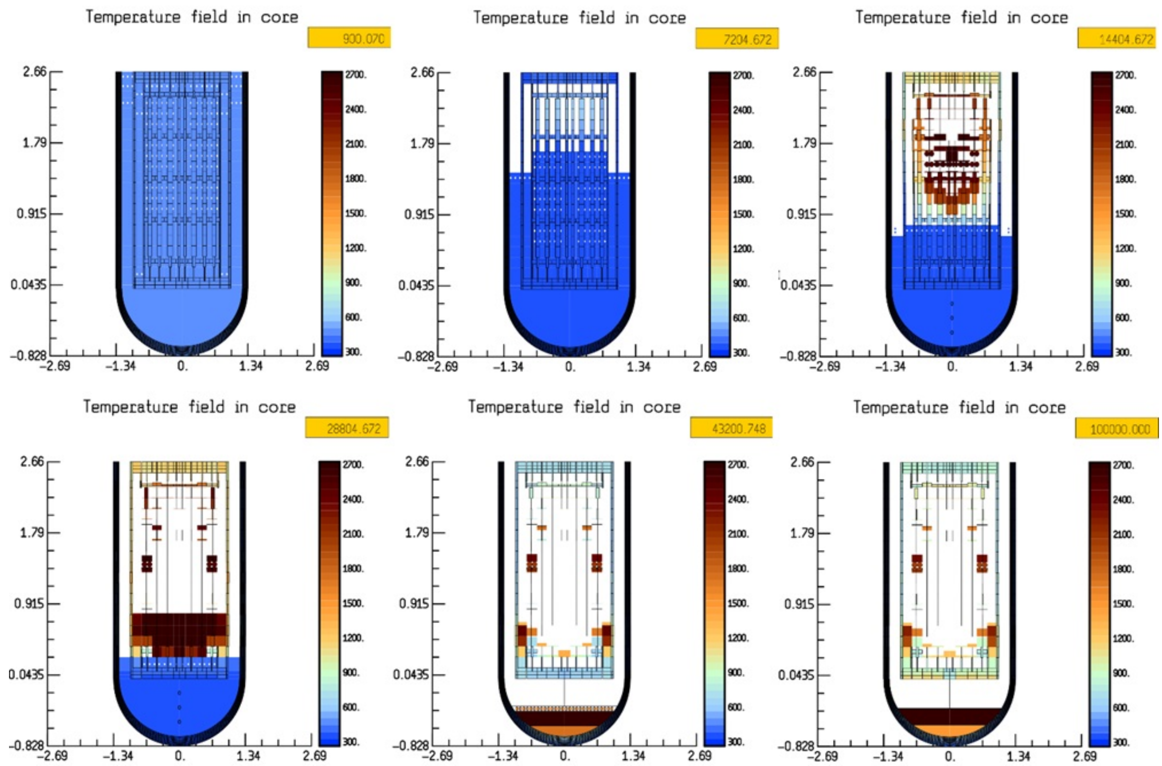

Figure 7: Schematic nodalization of the NC-iPWR Design model in ASTEC V3.1

of heavy metals and an upper layer of a mixture of oxide and metals coming from the core. As the event progresses, a layer of debris lies above the melts, which is then incorporated into them. By the end of the simulation no lower head failure was predicted.

5 CONCLUSIONS

The ASTEC code V3.1 was employed to investigate the progression of various hypothetical DBA and SA scenarios. The DBA study demonstrated that the code is capable of clearly reproducing the main thermohydraulic variables and the different phases of the progression of such scenarios, highlighting the effectiveness of the ECCS in continuing to cool the core through the opening of RVVs and RRVs. Additionally, the sensitivity analysis of the break size allowed for an examination of the thermohydraulic behaviour with breaks as low as 10% of the reference area. During the SA scenario, the code proved the ability to describe the accident progression from the initiating event through core degradation, the corium formation, and the relocation in the lower plenum. The total hydrogen production was $\sim$64 [kg], mainly due to the oxidation of zirconium and the molten pool. The amount of relocated corium was 5.8 [tons] for the analyzed case but no lower head failure was determined. In addition, the code allowed the description of the evolution of the different magma layers and the debris bed in the lower plenum. Finally, the evolution of the fission products from their release to the different system volumes was observed, capturing the different states in which the fission products are present in the containment and allowing for a detailed description. To further improve the understanding and assessment of nuclear safety in SMRs, additional hypothetical conditions should be tested.

ACKNOWLEDGEMENTS

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