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Iodine source term assessment under DBA SGTR accident scenario

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ABSTRACT

In the framework of the European project denominated Reduction of Radiological Consequences of design basis and design extension Accidents (R2CA), application of Best Estimate advanced codes for evaluating the radiological release under Design Basis Accident and Design Extended Conditions was considered. The aim is to assess the current modelling capabilities and to propose new calculation methodologies in order to produce more realistic evaluations of radioactive releases resulting from such accidents.

In this paper, the studies are conducted for a reference Design Basis Accident Steam Generator Tube Rupture scenario in a 3-loop PWR-1000 reactor in terms of key thermal–hydraulic and radiological variables that condition and characterize the fission product release to the environment. The assessment includes a comparative study of the results performed by three organisations using three different computational tools and approaches. The outcomes highlighted issues considered for the design basis evaluation, modelling differences as well as some challenges and limitations in carrying out such analysis.

1. Introduction

Nowadays, the assessment of the radiological consequences of a postulated accident in a nuclear power plant is still bounded by rigid assumptions and conservatism in the modelling (e.g. high containment leakage rates), which generally lead to a large overestimation of the radioactivity release. The EU H2020 Reduction of Radiological Consequences of design basis and extension Accidents (R2CA) project (Girault et al., 2022) aimed at developing new calculation methodologies and updated computer codes models in order to get more realistic evaluations of radioactive releases resulting from postulated Design Basis Accidents (DBA) and Design Extended Conditions (DEC). These new evaluations should serve to improve, on the one hand the emergency operating procedures (EOPs) and the severe accident management guidelines (SAMGs), and on the other hand, to propose new reactor systems instrumentation, thereby improving the safety of the plants. This is achieved through sharing and harmonizing good practices and methods used by the project partners. Indeed, several approaches are adopted by different countries in addressing and assessing the radiological consequences following DBA scenarios.

In this framework, Steam Generator Tube Rupture (SGTR) accident in a three loop Pressurized Water Reactor (PWR) is considered. This accident can affect the core integrity via the loss of primary inventory through the break and also lead to the inevitable release of radiation into the environment via the main steam safety valves or atmospheric dump valves (Bang et al., 2022). Generally, the adopted accident management strategy is to cool down the primary side via the intact SGs in order to reduce the break flow and consequently minimising the radioactivity releases. In the past, at least fourteen SGTR accidents were reported and all of them had single ruptured tube and were successfully mitigated without any damage to the reactor and without any significant release of radioactive material to the environment (Ullah et al., 2018). However, the followed methodology for performing DBA safety assessment of a SGTR case is generally based upon the consideration of conservatism at different levels of the analysis ranging from initial and boundary conditions, physical models and failure of the safety systems. Such approach may lead to an overestimation of the radioactivity releases, which sometimes could be too unrealistic. Thus, in order to get a more realistic estimation of the radioactivity releases, better estimate models with reduced conservatism could be considered.

In the current study, different approaches involving different conservatism levels and models of three different computer codes (Thermal-hydraulic system RELAP5 and CATHARE codes and severe accident MELCOR code) are considered. The objectives of the

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Table 1
Main operating conditions.

Component	Bel V	Tractebel	CIEMAT
Reactor power	1000 MWe	1000 MWe	1000 MWe
Primary side pressure:	15 MPa	15.5 MPa	15.5 MPa
Secondary side pressure:	7 MPa	7.3 MPa	6.98 MPa
High pressure injection system (HPSI) setpoint.	P < 11 MPa	P < 12 MPa	P < 12 MPa
Low pressure injection system (LPSI) setpoint.	P < 2.5 MPa	P < 2.2 MPa	P < 2.5 MPa
Accumulators' setpoint.	P < 4.5 MPa	P < 4.5 MPa	P < 4.5 MPa

Table 2
Main sequence of events.

Event (Tn)	Description
T1	Steady-state hot full powerDouble ended break at the apex position of the longest U-tube
T2	Control volume charge flow and pressurizer heater are activated.
T3	Low pressurizer level reached, letdown and heaters are disabled
T4	SRAM Signal (Low pressurizer pressure or manually)
T5	Reactor trip, MFW stopped MSIVs closed, AFW to all SGs
T6	SGs relief valves open RV of SG3 remains stuck open
T7	High-pressure safety injection setpoint is reached.
T8	Operator identifies and stops the AFW toward the affected SG.
T9	Operator isolates the affected SG and closes the faulted RV, and the stop of the radioactive release.

comparative study are, on the one hand, to highlight the main features and models of each code in performing assessment of the considered SGTR scenario and, on the other hand, to demonstrate that unnecessary conservatism in the assumptions could be reduced.

2. Plant and accident scenario description

The reference reactor is a 3-loop Pressurized nuclear Water Reactor (PWR) of 1000 MWe. The Reactor Coolant System (RCS) is mainly composed by the Reactor Pressure Vessel (RPV), a pressuriser, three cooling loops having each a circulating pump and a Steam Generator (SG) containing thousands of vertical U-tubes. Table 1 outlines the main operating initial conditions as well as the set points related to the safety injection systems (Bradt, 2019).

The transient starts at the occurrence of a double-ended break located in the apex of the longest U-tube of one of the three SGs. Generally, at this elevation the U-tubes are subjected to the highest mechanical stresses and therefore are more subjected to fatigue and rupture. On the other hand, for this kind of scenario, the radioactivity release is maximised as the break remains uncovered for a longer period of time with respect to the lower part of the U-tubes.

Table 2, shows the sequence of events of the considered SGTR scenario. After the opening of the break (event T1), the primary coolant inventory, the pressure and the water level at the pressurizer, decrease. The control systems automatically get activated to restore and maintain both the pressurizer pressure and level (event T2). However, due to the continuous primary inventory loss, the makeup system will not able to compensate the primary to secondary break flow, and therefore pressure and pressurizer level keep on decreasing slowly. These conditions will lead, shortly after, to the isolation of the Chemical and Volume Control System (CVCS) letdown line and the pressurizer heaters due to the low pressurizer level (event T3). The SCRAM signal, will be either triggered by the low pressurizer pressure signal or manually by the operator action after 30 min from the opening of the break (event T4). As a result of the reactor trip, both the CVCS charge line and the Main Feedwater (MFW) systems are isolated. The turbine is tripped, the Main Steam Isolation Valves (MSIVs) are closed and the Auxiliary feedwater (AFW) system is activated (event T5). The accident scenario also assumes that the condenser steam dump is not available. This causes a rise of the secondary pressure and the opening of the Relief Valves (RVs) in all the SGs (event T6). For the current DBA scenario, it is assumed, as a single failure, that the RV in the affected SG remains stuck fully open until it is manually closed by the operator. The objective is to maximize the radioactivity release to the environment.

Following the EOPs, the operator identifies the faulted SG and stops the AFW toward the affected SG (event T8). Subsequently, the affected SG is isolated and the faulted RV is closed (event T9). This last operator action put an end to the radiological releases from the affected SG.

3. Comparative study

3.1. BEL V: CATHARE model & hypothesis

The DBA scenario is numerically simulated using the advanced best-estimate fully implicit thermal–hydraulic system code CATHARE2 (Darona, 2018) developed by CEA, EDF, AREVA, and IRSN. The adopted nodalization is based on a fully 1-D model, including all the 3 primary cooling loops and their associated safety injection systems as well as detailed representation of the RPV and the secondary side systems. Particular attention is given to the SG nodalization where three U-tubes

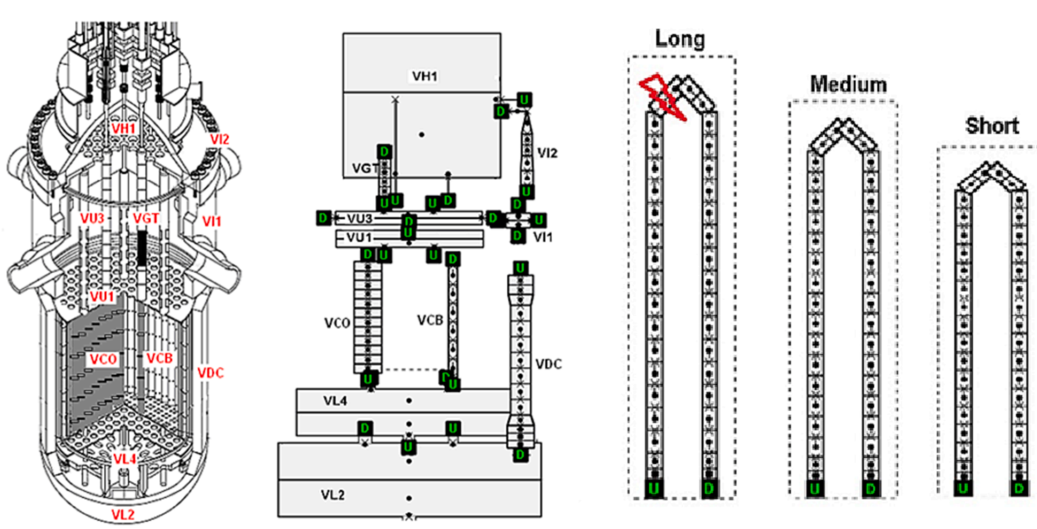

Fig. 1. CATHARE RPV modelling and SG U-tube representation.

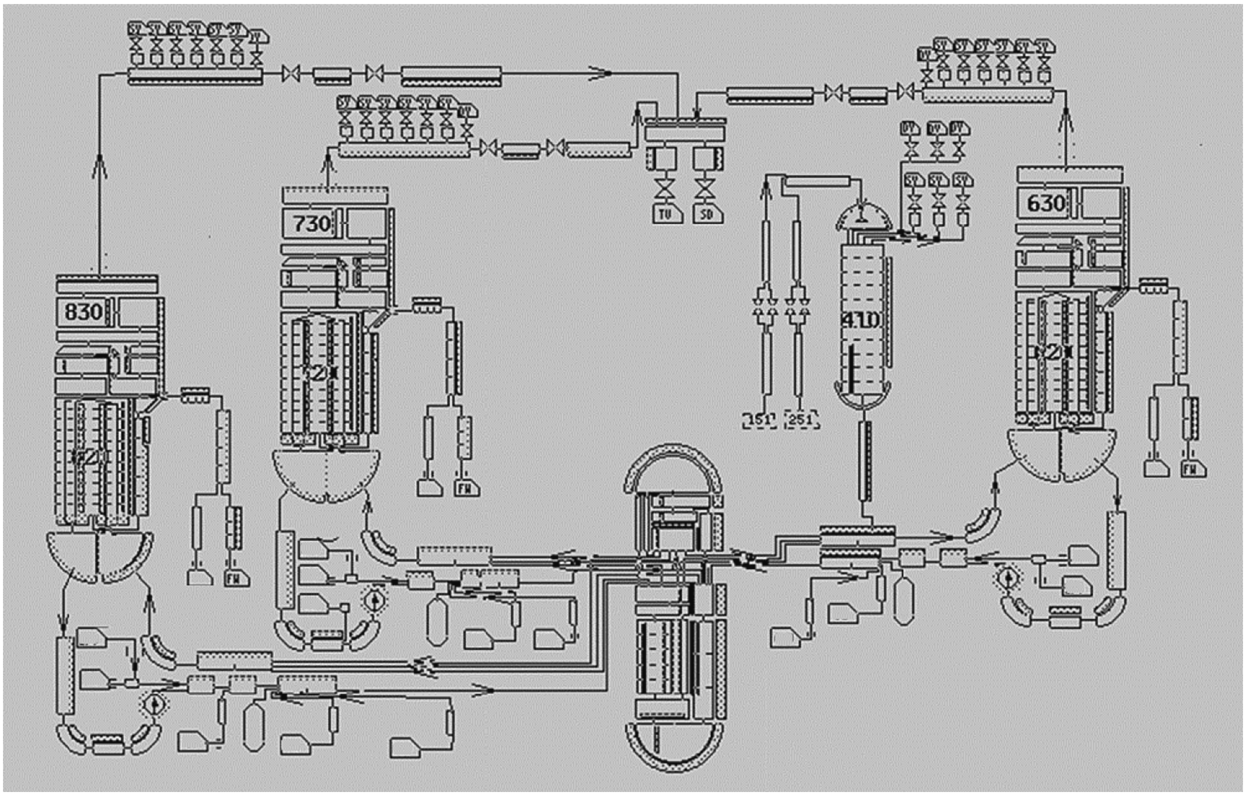

Fig. 2. TH RELAP model adopted by Tractebel.

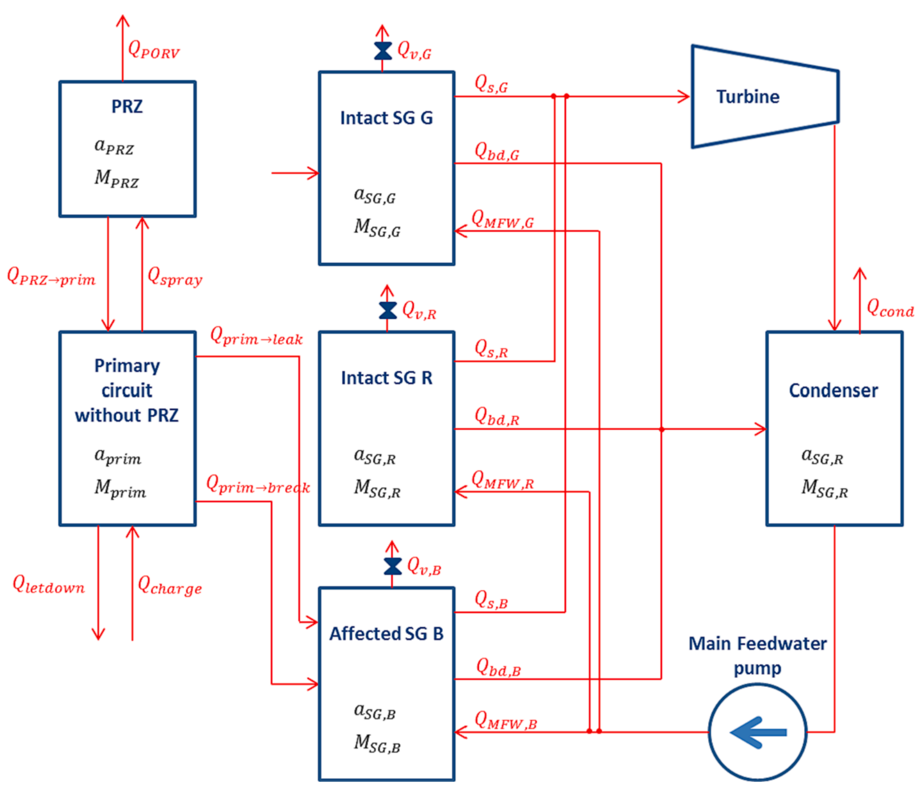

Fig. 3. Illustration of the Tractebel FP transport model with its six different components.

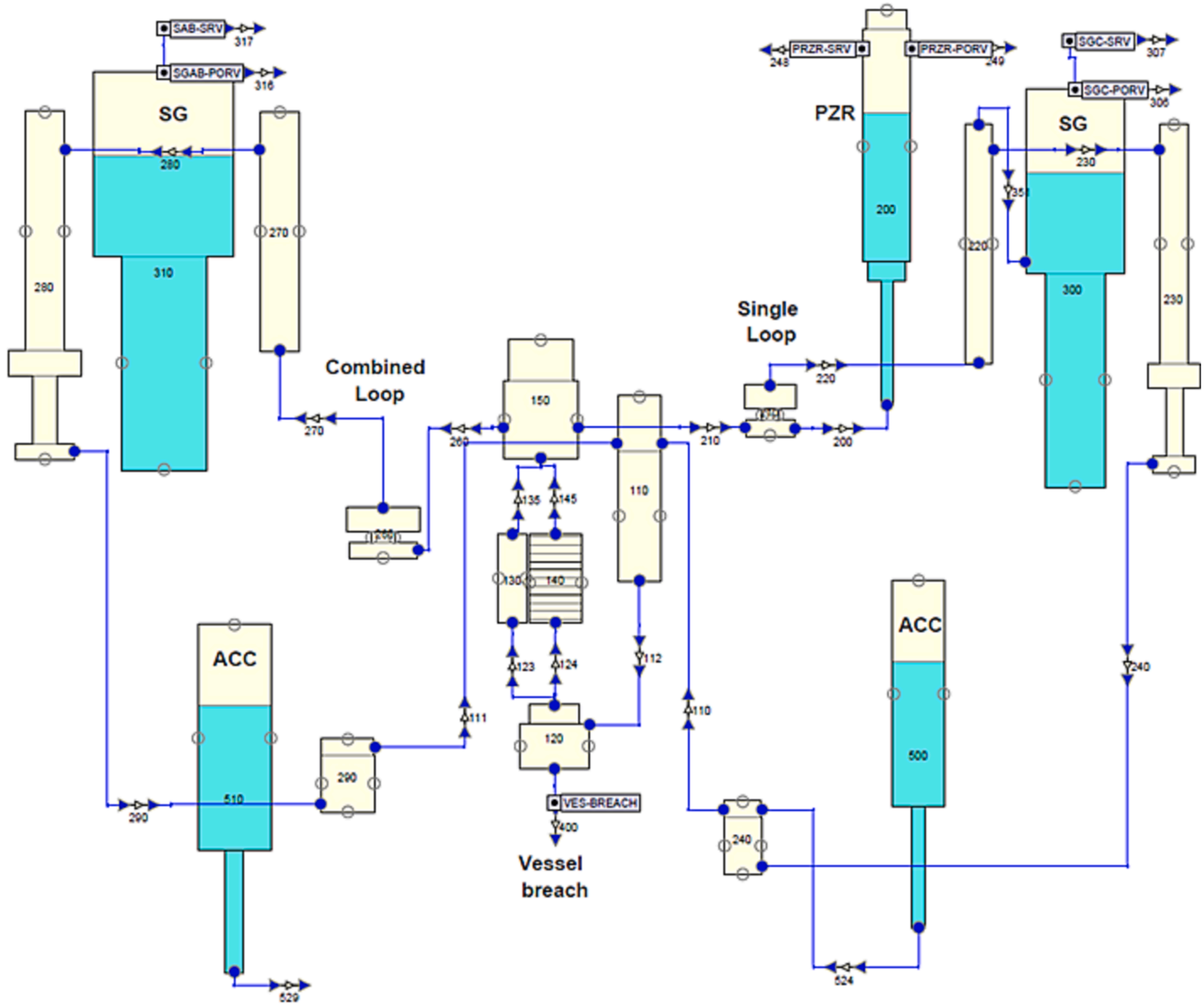

Fig. 4. RCS and RVP nodalization adopted by CIEMAT.

lengths (short, medium and long) with detailed ascending and descending nodes are considered (see Fig. 1). This kind of nodalization is more suitable for the current study, since the break is supposed to take place at the apex of the longest U-tubes. The goal is to maximise the time period of the break uncovery.

The radioactivity transport is simulated using the CATHARE code transport model coupled with the TH calculations. However, for the calculation of the radioactivity release into environment a number of assumptions is considered as the following:

- Radioactivity release mechanism due to flashing, partitioning and atomisation are considered.
- Only the Iodine 131 (^{131}I) is considered for the current calculations even though other isotopes could be handled by the code together with the ^{131}I.
- The iodine release due to the spiking was modelled as the following:
• A radioactive source was modelled at the core level.
• The source injects high concentrated ^{131}I with a very low and coolant mass flow rate (~10^{-3} kg/s) to avoid disturbing the heat and mass balance in the core.
• The ^{131}I concentration is varied according to a predefined spiking release rate.
- No radioactivity is considered in the affected SG at the beginning of the accident.
- The flow through the SG tube break is considered isenthalpic.
- The radioactivity release through the partitioning effect is considered using the constant partitioning coefficient (PC = 100) between the primary liquid activity and the steam flow through the stuck open relief valve by subtracting the steam released by flashing.
- The radioactivity release by atomisation and flashing is assumed to be 100 % released into environment, independently of the SG relief valve flow rate.

3.2. *Tractebel: RELAP5 model & hypothesis*

Tractebel conducted the study using the best-estimate semi-implicit TH code RELAP5 mod 2.5 (Ransom, 1985), along with a fission product (FP) transport model developed by (Van Hove et al., 1997) to estimate radiological consequences.

The reference three loop PWR 1000 MWe was described using a rather detailed meshing scheme (see Fig. 2). The TH part of the vessel is subdivided into 21 control volumes. The rest of the RCS is subdivided in 134 control volumes: 12*3 for the cold/hot legs and the u-branches for the 3 loops, 26*3 for the primary side in the SGs, 20 control volume for the pressurizer and its surge line and 6 control volumes for the broken

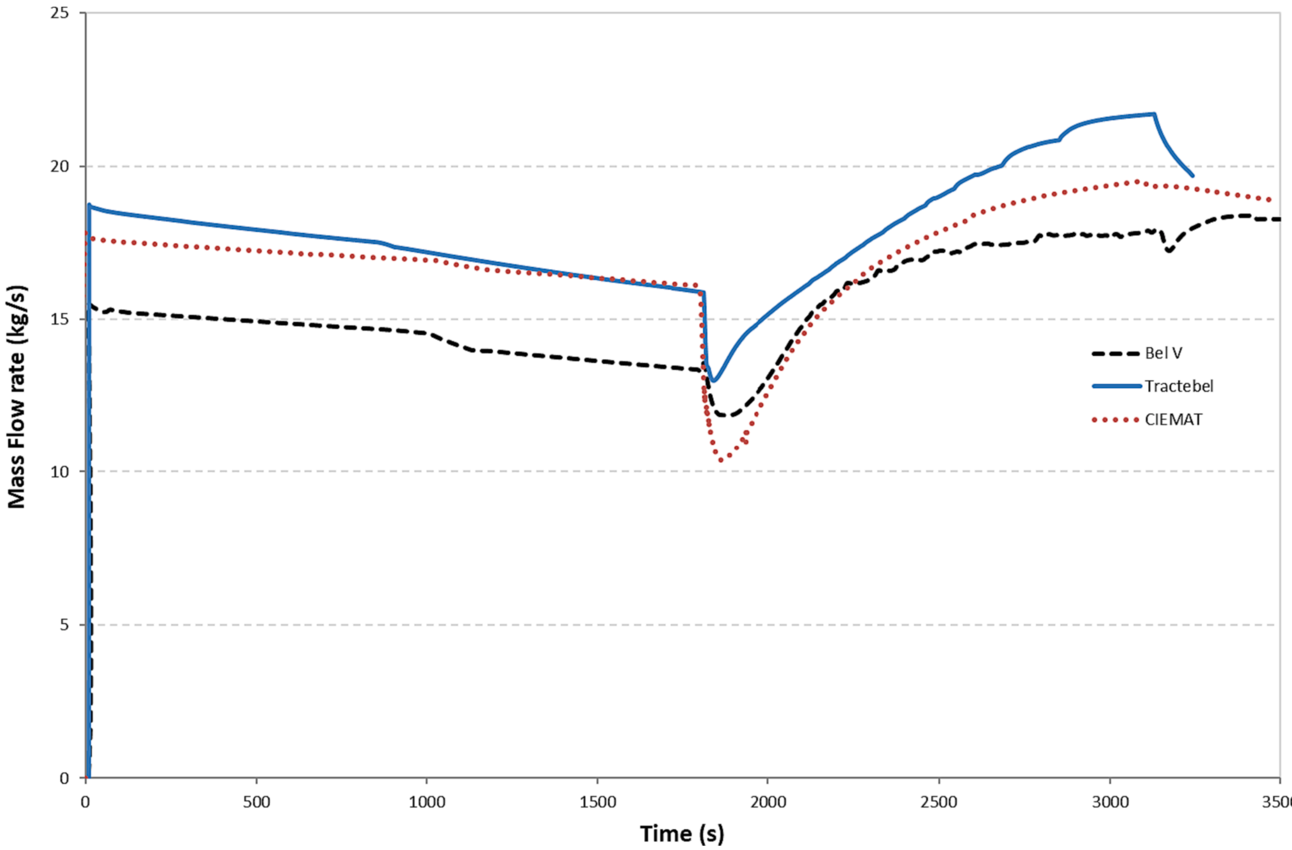

Fig. 5. SGTR total break mass flow rate.

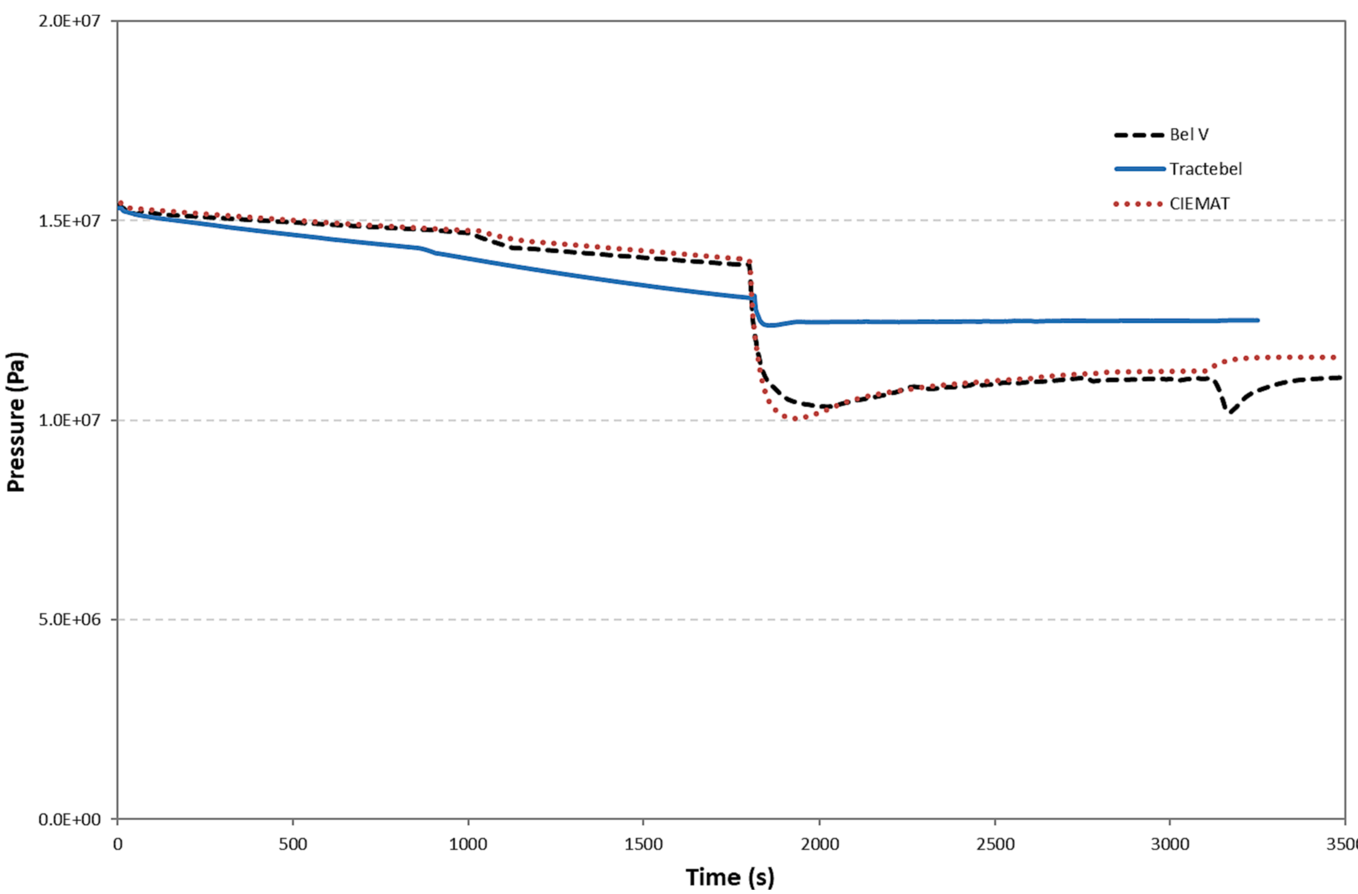

Fig. 6. Primary pressure.

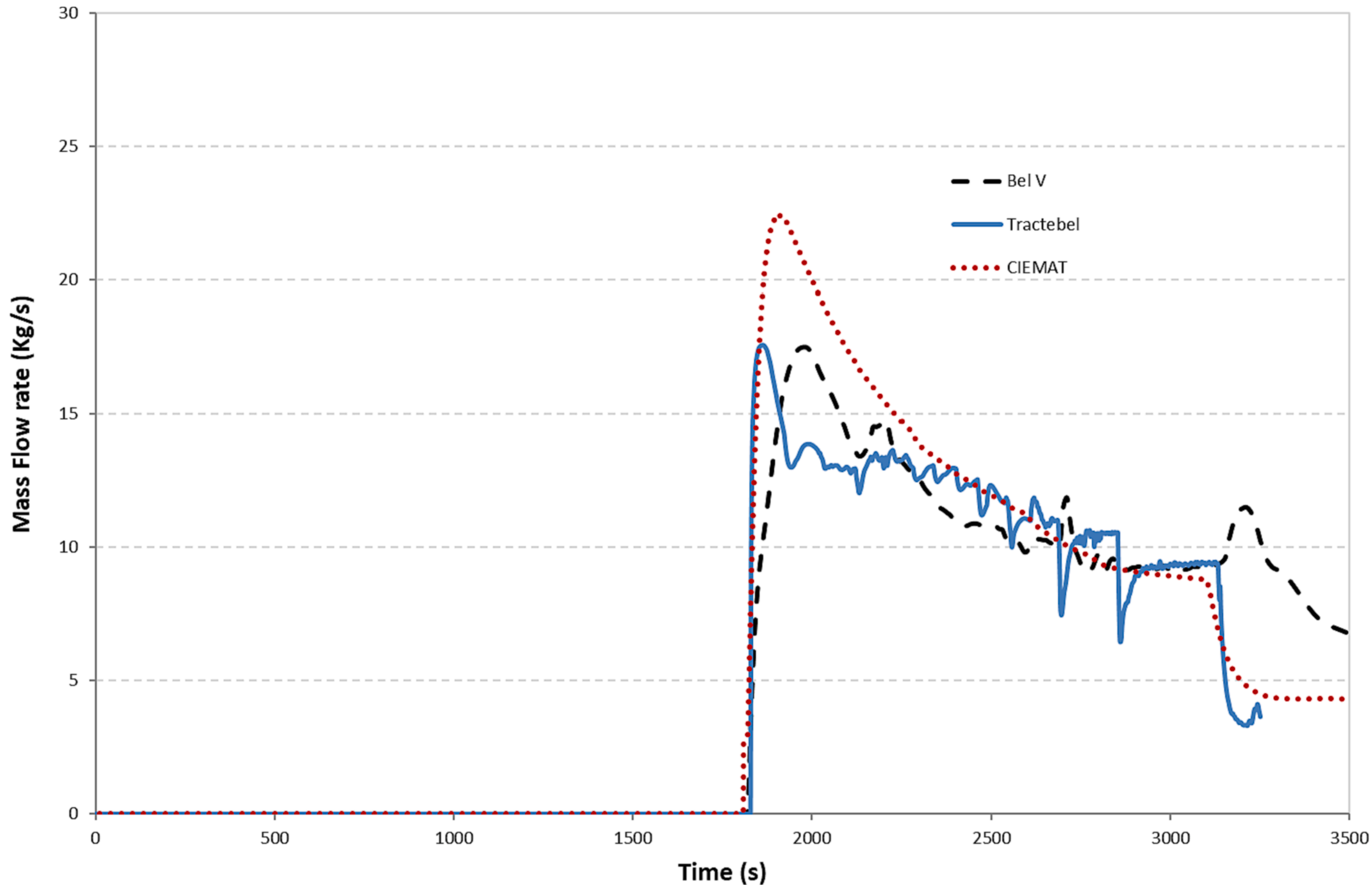

Fig. 7. HPSI mass flow rate.

tubes in the affected SG (SG-1). Finally, the secondary side in the SGs is subdivided in 32 control volumes. The RELAP5 system model is able to simulate all operator actions, which are consistent with the Emergency Operating Procedures (EOP), as well as the actual physical evolution of the plant parameters.

As mentioned above, to perform a more accurate assessment of the radiological consequences, the RELAP system deck was coupled with a specific model reproducing the transport of FPs notably ^{131}I in the plant.

The FP transport model adopts a simplified plant nodalization of 6 components (as illustrated in Fig. 3) to track the amount of ^{131}I in the primary circuit, the pressurizer, each of the SGs and the condenser. It solves six (one for each component) coupled first-order ordinary differential equations via RELAP control variables to calculate the balance of iodine activity in the plant.

The set of differential equation uses the TH results predicted by the RELAP system model as input to simulate the transport of iodine between components and its release into the environment. Coupling is performed at each time step within RELAP5 to utilize the most update TH data in the calculation of the FP transport.

The main phenomena of iodine production, retention and release described by the FP transport model are:

- **Spiking** (release of iodine from defect fuel rod): This phenomenon refers to a sudden increase in the concentration of iodine activity in the primary coolant following changes in reactor power or reductions in RCS pressure. Spiking is modelled as a source of iodine proportional to the initial contamination in the RCS, so that it is a consequence of the same number of fuel defects.
- **Partitioning** (partial transfer of iodine present in liquid water to the gas phase): A part of the iodine dissolved in the liquid water within the SG is transferred to the steam phase (the value of the partitioning coefficient is set to 100);
- **Flashing** (instantaneous vaporization of primary coolant): In case the primary coolant is superheated with respect to the secondary side (due to the pressure difference), a fraction x of the break flow rate will vaporize instantaneously by an isenthalpic process called flashing. In this case the specific activity of the flashed steam is assumed equal to the primary specific activity;
- **Atomisation** (entrainment of primary coolant droplets): In case of flashing of a part of the break flow, the liquid fraction of the break flow breaks up in mist of primary coolant droplets. If the break is uncovered these droplets are entrained with the flashed steam and a fraction y of them will be able to by-pass the SG separators and dryers. As for the flashing phenomenon, the specific activity of the entrained droplets is set equal to the primary specific activity;
- **Moisture carry-over** (entrainment of secondary water droplets): Due to the bursting of steam bubbles at the water surface, secondary water droplets are 'thrown' in the steam phase and entrained. However, these droplets have much bigger dimensions than in case of atomisation. Therefore only a very small fraction of these droplets is able to by-pass the SG separators and dryers;
- **Dry-out**: In case the affected SG completely dries out, it is conservatively assumed that all activity in the liquid phase is released to the environment.

3.3. CIEMAT: MELCOR model & hypothesis

The integral code MELCOR 2.2 release 18019, developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission, is an engineering-level computer code that models the progression of severe accidents in nuclear power plants (Humphries et al., 2021). In

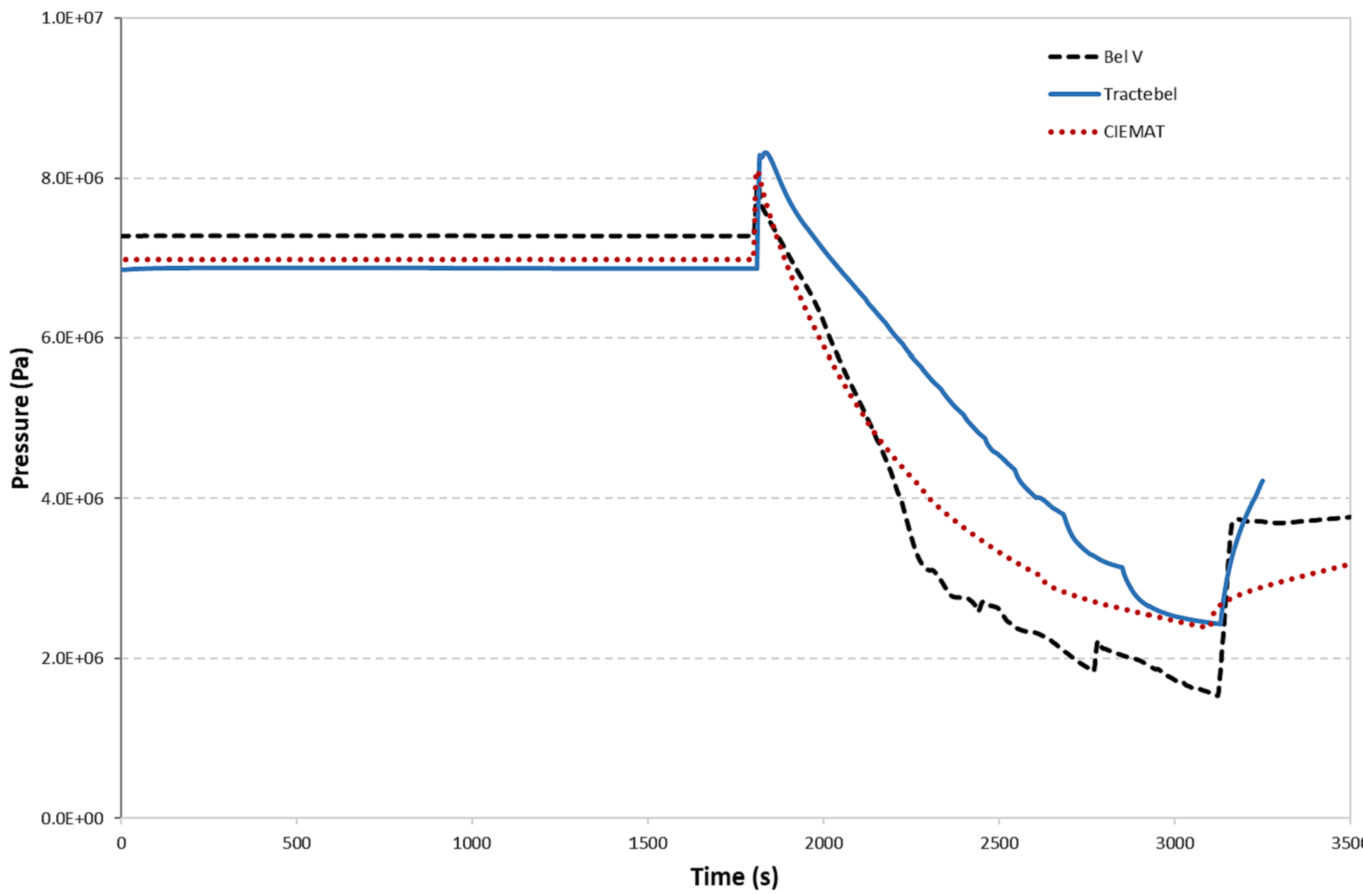

Fig. 8. Secondary pressure evolution in the affected SG.

particular, a broad spectrum of accident phenomena in both boiling and pressurized water reactors is treated in MELCOR in a unified framework (Humphries et al., 2021). The Radionuclide (RN) package calculates the release and transport behaviour of fission product vapours, aerosols and other trace species, including releases from fuel and debris, aerosol dynamics with vapour condensation and re-vaporization, deposition on surfaces, transport through flow paths, and removal by engineered safety features. The RN package operates on the basis of material classes, which are groups of elements with similar chemical properties. The radionuclide initial inventory is based on ORIGEN code results (Humphries et al., 2021).

The simplified model of the reference NPP model is used. It consists 32 control volumes (CVs), 46 flow paths and 66 heat structures. The RCS (28 CVs) is modelled by two loops (see Fig. 4), one of which combines the two intact coolant loops, and the other one representing in which the Pressurizer and the failed SG are included. The core region is divided in 4 radial rings and 15 axial nodes. The first three rings represent the fuel and the fourth ring represents the core bypass region. The AFW and HPSI safety systems are modeled through MELCOR control functions.

The SG tube rupture (primary-to-secondary system connection) is modeled by a flow path located at the top of the inverted-U tube bundle, which cross-section is double the one of a single SG tube. The failed SG-RV valve connecting SG and environment is modeled by a closed flow path which opening is regulated by a control valve (a MELCOR control function).

In the analysis of the FP behaviour the following key assumptions were made (Humphries et al., 2021), (NUREG-800, 1981):

- Initial ^{131}I activity concentration in the coolant of 1 μCi/g DE ^{131}I.
- Initial noble gases activity concentration of 1 μCi/g DE ^{131}I.
- An iodine spiking phenomenon right at the reactor trip.
- The spiking model assumes that the iodine release rate from the fuel to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value stated in the NPP Technical Specifications (see Fig. 9). Note that this assumption does not lead to a "spike"; instead, it assumes a continuous and monotonous iodine accumulation until the event is terminated. This would become a bounding case of iodine spike (Reisi Fard, 2011)
- No radioactivity is considered in the secondary side of the affected SG at the beginning of the accident.
- The three distinct mechanisms for transport of the iodine to the environment, i.e. flashing, atomization and partitioning were modelled with the application of the flashing and fog formation model in the case of pool entering a volume through a flow path (Humphries et al., 2021) using MELCOR control functions. The iodine partitioning between aqueous and gas phases was considered by setting a constant partitioning coefficient (PC = 100) in the affected SG. Other fractions of gas radioactivity stems from the primary water atomisation and flashing in the gas phase of the affected SG.

4. Results & discussions

The numerical simulation results of the DBA SGTR assessment, using three different computer tools and hypothesis are outlined hereafter.

4.1. BEL V Calculation results

The steady state calculation were performed for a pseudo transient regime of 10000 s during which the main key parameters reach their stabilized values. The transient calculations were performed for a time

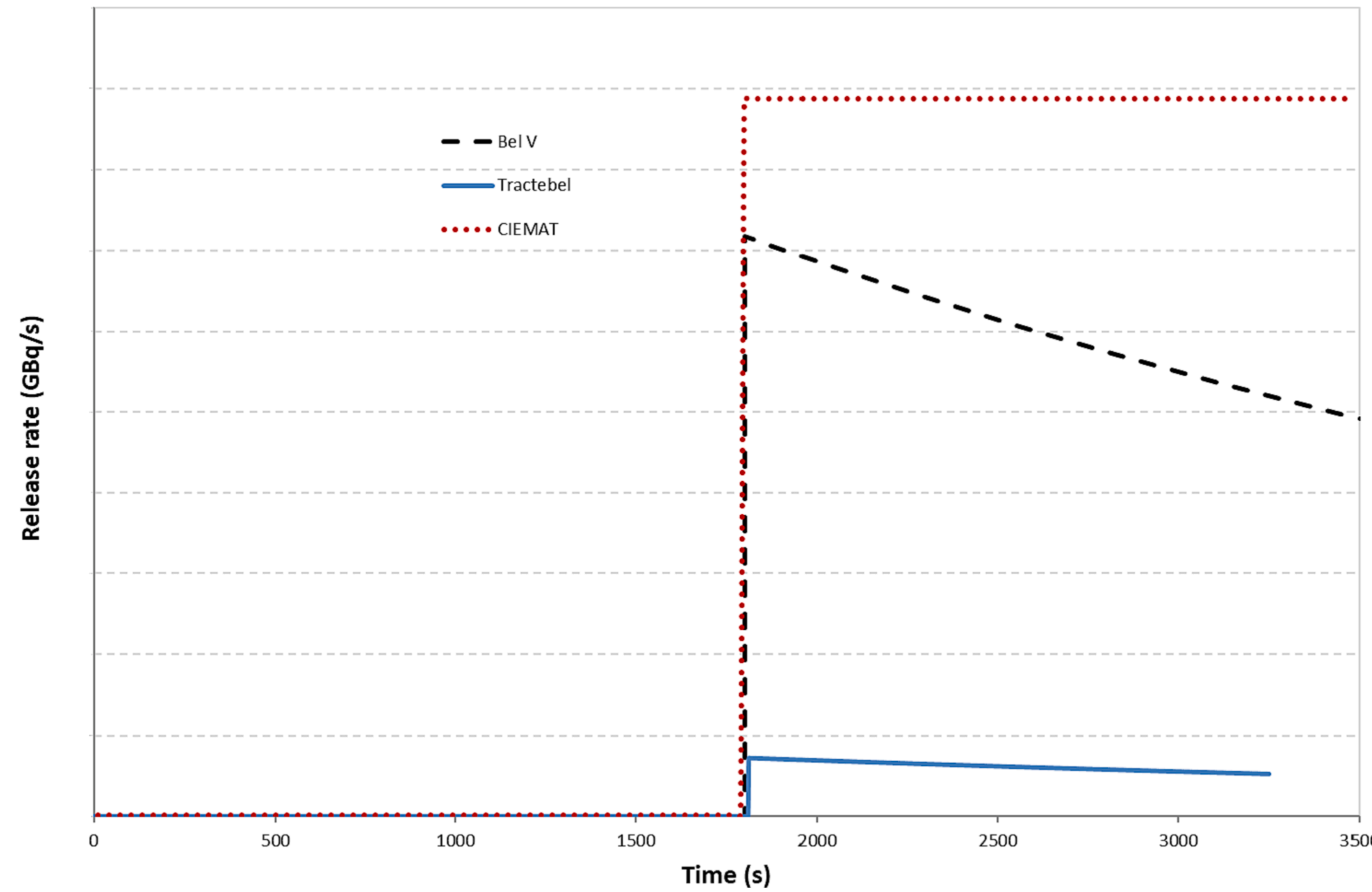

Fig. 9. Iodine activity released by the spiking phenomenon.

period of 3500 s starting from break time occurrence, and ending shortly after the time in which the operator closes the stuck open Relief Valve (RV). Fig. 5 shows the total break flow coming from the cold and hot side of the broken U-tube. The primary pressure decrease and evolution is shown in Fig. 6. In this calculation, the SCRAM setpoint by low primary pressure is not reached, and a manual SCRAM is activated by the operator action 30 min after the beginning of the transient. The primary side pressure decrease stops afterward due to the activation of the high-pressure safety injection system (HPSI). This leads to a stabilization of the primary pressure around the maximum head of the HPSI. The injected HPSI mass flow rate is shown in Fig. 7.

The secondary pressure evolution in the SG with broken U-tube (SG-3) is shown in Fig. 8. After a short rise at the beginning of the transient the pressure in SG-3 decrease and resume later after the closure of the stuck open RV. The abrupt power change together with the primary pressure decrease, following the SCRAM, lead to the spiking phenomenon, which results into a significant increase of the primary side activity (see Fig. 9). This leads to a faster increase of the activity in the secondary side in SG-3. Fig. 10 shows the amount of steam flow through the stuck open RV. The total predicted radioactivity release by the three airborne mechanisms is shown in Fig. 11. The latter is the result of three contributions where the radioactive release by atomisation appears to be the dominant mechanism (70 %), followed by the flashing (17 %) and partitioning (13 %) mechanisms.

4.2. TRACTEBEL calculation results

Prior to performing the SGTR transient calculations, a steady-state calculation lasting 300 s was conducted to verify that calculated initial conditions were representative of the actual plant operating conditions. At transient initiation, t = 10.0 s, a double-ended break of one U-tube at the top of the U-bundle of the affected SG is postulated to occur. The iodine initial specific activity in the primary circuit is set at 1 GBq ^{131}I per ton of coolant.

The initial total primary-to-secondary break flow rate through the SGTR is about 18.7 kg/s (Fig. 5). Due to the loss of primary coolant, the level and the pressure in the pressurizer (PZR) decrease. Several control systems intervene to maintain the PZR pressure (increase of PZR heaters electrical power) and level (increase of the CVCS charge flow rate) at this stage. Despite the automatic actions aiming at maintaining the PZR level and the pressure, both continue to decrease (Fig. 6). This eventually leads to the isolation of the PZR heaters and CVCS letdown line on low PRZ level signal.

The isolation of the letdown line (around 900 s), slows down the lowering of the PZR level. Nevertheless, it continues decreasing as CVCS charging flow is not able to compensate the SGTR break flow. The water level in the affected SG is kept constant reducing the refrigerant injection of MFW (automatic control). 1800 s after the initiating event, it is assumed that SCRAM is manually triggered by operators (consistently with emergency operating procedure).

The SCRAM combined to the isolation of MFW lead to a collapse of the SGs water level. The resulting closure of the MSIVs causes a pressure build-up in the secondary side of all SGs, triggering the SG Relief Valves (RVs) (see Fig. 10). In the case of the SG-1, the RV remains stuck open, initiating the release of contaminated coolant into the environment. The RV stuck open accelerates depressurization of the affected SG (Fig. 8),

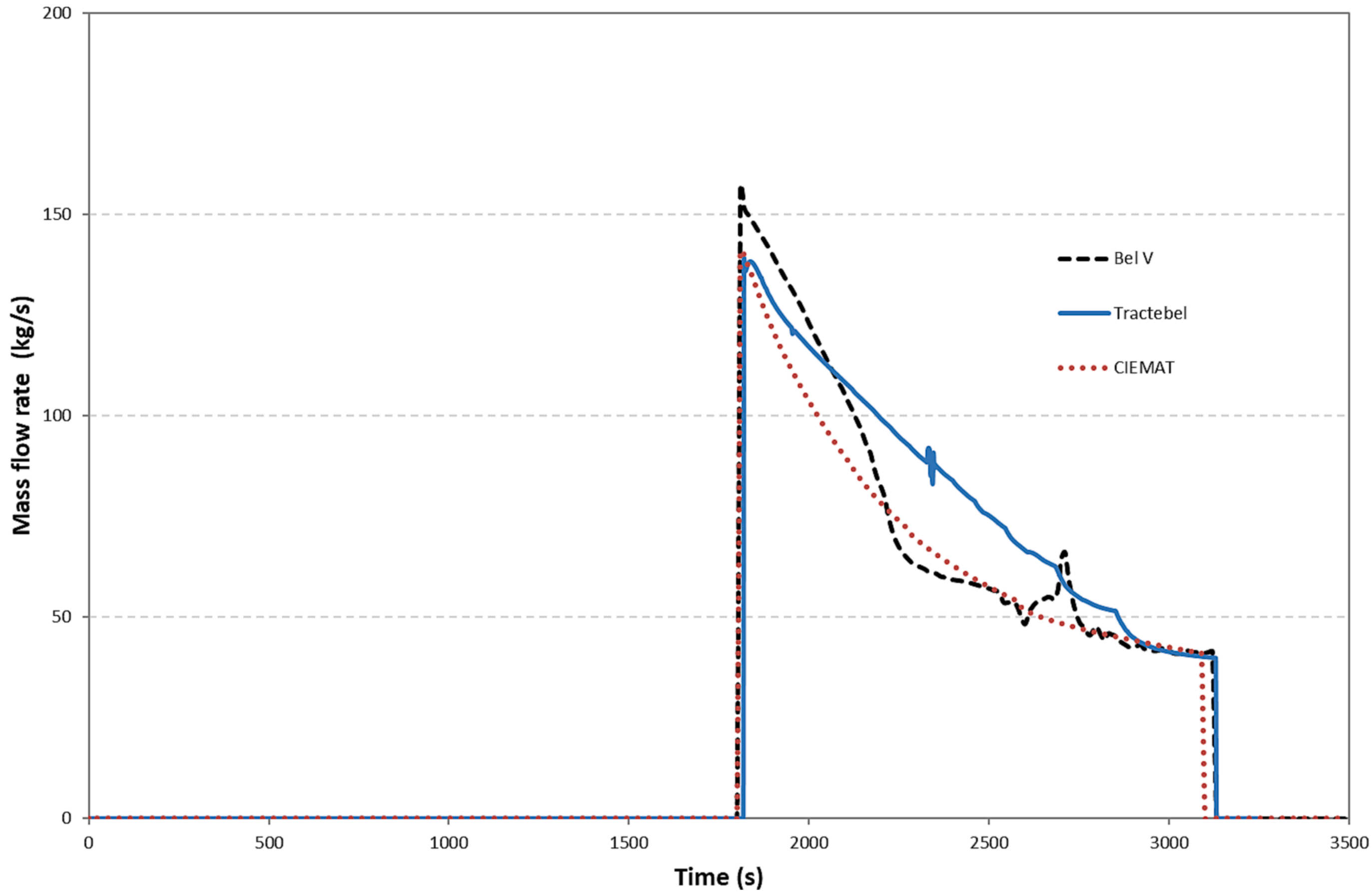

Fig. 10. Total steam flow through the stuck open relive valve.

leading to an increase in the primary-to-secondary coolant flow through the SGTR (Fig. 5). Despite this, the water entering the shell side of affected SG as a result of AFW injection and through the SGTR cannot compensate for the steam released through the stuck open RV. In fact, the persistence of these conditions causes a reduction of the water mass in the SG up to the 10 % of the initial inventory at 3000 s. At around 1320 s after SCRAM, the operators isolate the affected SG by stopping the AFW flow and closing the RV. The transient simulation ends when the release through the affected SG RV ceases completely around 3200 s. Prior to reactor trip the specific activities in the various components of the FP transport model remain quite low and nearly constant, except in the affected SG, where the activity rises due to the SGTR break flow. The 'iodine spiking' occurring as a result of the SCRAM at 1800 s generates a sharp increase in the activity level within the primary circuit and to a lesser extent in the damaged SG (see Fig. 9). The SG RV stuck open causes a large part of the primary circuit activity to flow into damaged SG through the breach. In addition, as the water level in the affected SG continues to decrease, the height of liquid above the break is not covered anymore by water and iodine release through the atomisation mechanism occurs. The prolongation of these conditions over time (SG water inventory reduction) produces a steady acceleration of flashing and partitioning iodine release rates. The total radiological releases from affected SG is shown in Fig. 11. It is the sum of different radioactive contributions of the partitioning, the flashing and the atomisation phenomena. The latter constitutes 62 % of the total release followed by the partitioning (28 %) and the flashing (10 %).

4.3. CIEMAT calculation results

Initial calculations were executed during 10000 s to obtain the required steady state conditions before the start of the tube rupture in the affected SG. The rupture of one tube of the affected SG is postulated to occur at time t = 0.0 s. Fig. 5 shows the mass flow rate through the ruptured SG tube. Due to the primary to secondary coolant leakage, the primary pressure (see Fig. 6) decreases. Taking into account that the signal of low primary pressure for the reactor trip is not reached, the reactor trip is produced manually by operator after 30 min from the beginning of the SG tube rupture. The primary pressure decrease is stopped once the HPSI is activated (see Fig. 6).

Fig. 8 shows the evolution of the pressure in the affected SG. At the beginning of the transient, the SGs pressure increase rapidly causing the opening of the RV and steam discharge to the atmosphere. A sharp depressurization occurs in the affected SG because the assumed stuck open RV in the ruptured SG. This decrease continues until the closure of the RV by operator action.

The iodine spiking produced as a consequence of power and pressure changes, is postulated to occur coincident with the reactor trip (1800 s after the initiating event), leading to the increased released activity to the RCS and therefore substantial increase of activity in the affected SG as shown in Fig. 9. Fig. 10 shows the steam flow through the SGs RV and Fig. 11 shows the amount of total radioactivity released through the stuck open RV considering the three mechanisms of iodine transport. The predominant mechanism is the flashing (86 %), followed by the partitioning (13 %) and the atomisation (1 %). The radioactive release is stopped at 3100 s once the stuck RV is closed by operator.

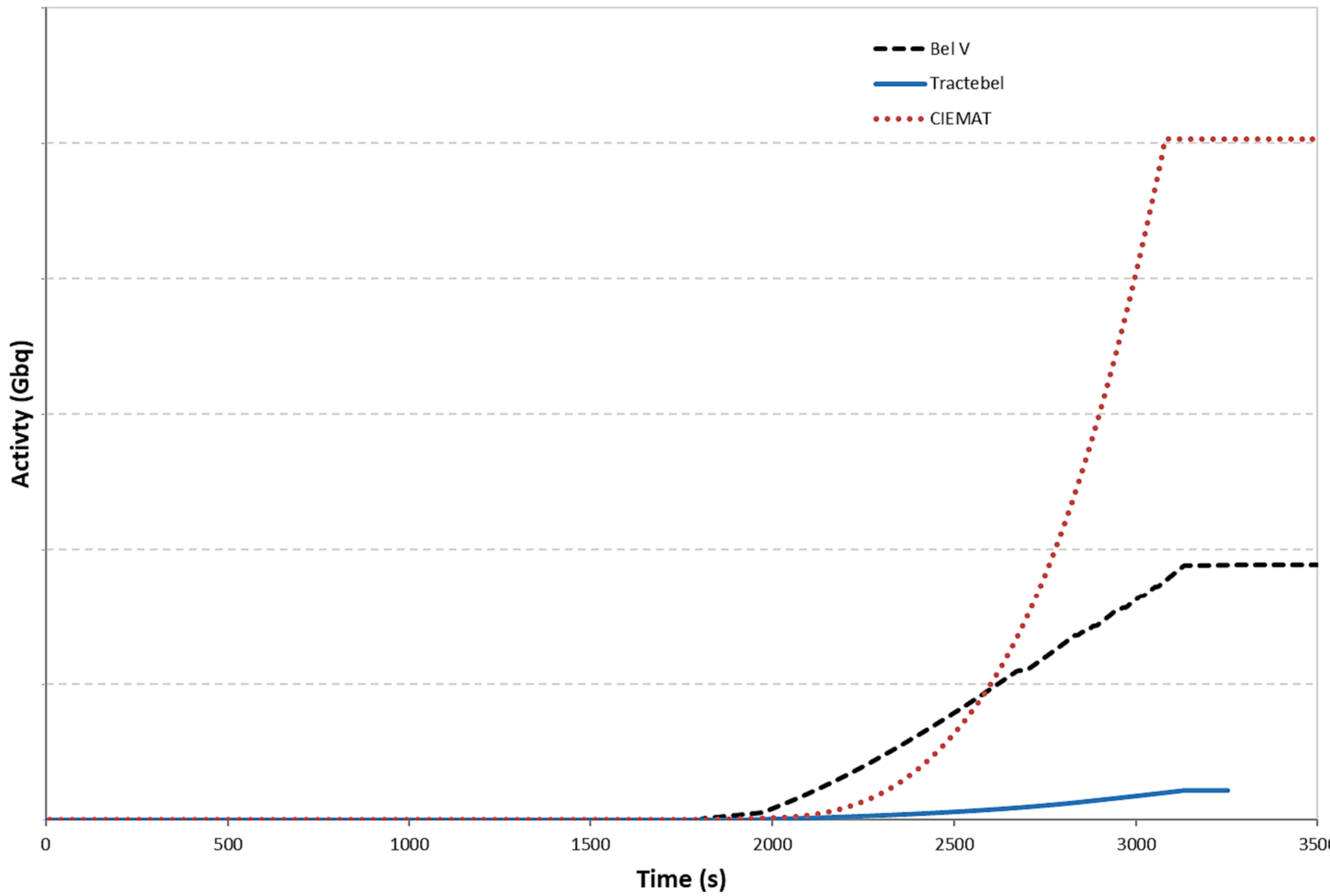

Fig. 11. Total released activity to the environment.

As could be seen in Fig. 11, CIEMAT model predicts the larger amount of the radiological release from the affected SG. This is due to the use of a more conservative iodine spiking model (see Fig. 9) in accordance with NUREG-800.

A sensitivity analysis of the MELCOR modelling to the water drops size in the atomization model has been conducted. As shown in Fig. 12, the droplet size affects drastically the relative importance of each phenomenon. If a droplet size between 35 and 40 μm had been assumed instead of 10 μm, atomization might have been brought to roughly the same values as in the other predictions.

5. Summary & conclusions

In this paper, a DBA scenario related to the SGTR accident in a three loop 1000MWe PWR is considered aiming at assessing the radioactivity release to the environment using different codes and hypothesis. Two thermal–hydraulic system codes, namely CATHARE and RELAP5 as well as the severe accidents MELCOR code were used. In addition, specific hypothesis and models were adopted mainly to overcome some code modelling limitations in simulating key physical phenomena that govern the radioactivity transport like the atomisation, the isenthalpic conditions through the break and the radioactivity repartition in liquid and vapour phase.

The assessment study shows, on the one hand, that the used codes, notwithstanding their numerical and modelling differences, predict similar evolution of the thermal–hydraulic key parameters. On the other hand, large difference is obtained with respect to the prediction of the amount of released radioactivity. This is mainly due to differences in the used models and hypothesis that have been adopted by the three organizations to predict the spiking and the radioactivity transport by each contributor mechanism. Actually, CIEMAT used the most conservative hypothesis related to the spiking model and each organisation used its own conservative hypotheses to evaluate the amount of radioactivity resulting from the three main contributors, i.e. atomisation partitioning and flashing. For CIEMAT, the main contributor of the radioactivity release is the flashing (86 %) followed by the partitioning (13 %) while in the Bel V and Tractebel case, the predominant contributor comes from the atomisation phenomenon (70 % and 60 %, respectively). On the other hand, difference with respect to the second contributor is observed; it is the partitioning for Tractebel (28 %) and CIEMAT (13 %) and the flashing (17 %) for Bel V.

The observed differences in the prediction of the radioactivity releases highlights the need to harmonise approaches, particularly those related to radioactivity transport, and no less important to unnecessary conservatism in the assumptions made. This could be achieved by the use of more advanced modelling and Best Estimates approaches that allow getting more realistic evaluations of radioactive releases.

CRediT authorship contribution statement

A. Bousbia Salah: Writing – original draft, Methodology, Formal analysis, Writing – review & editing. **M. Di Giuli:** Writing – original draft, Methodology, Formal analysis. **P. Foucaud:** Writing – original draft, Methodology, Formal analysis. **R. Iglesias:** Writing – original

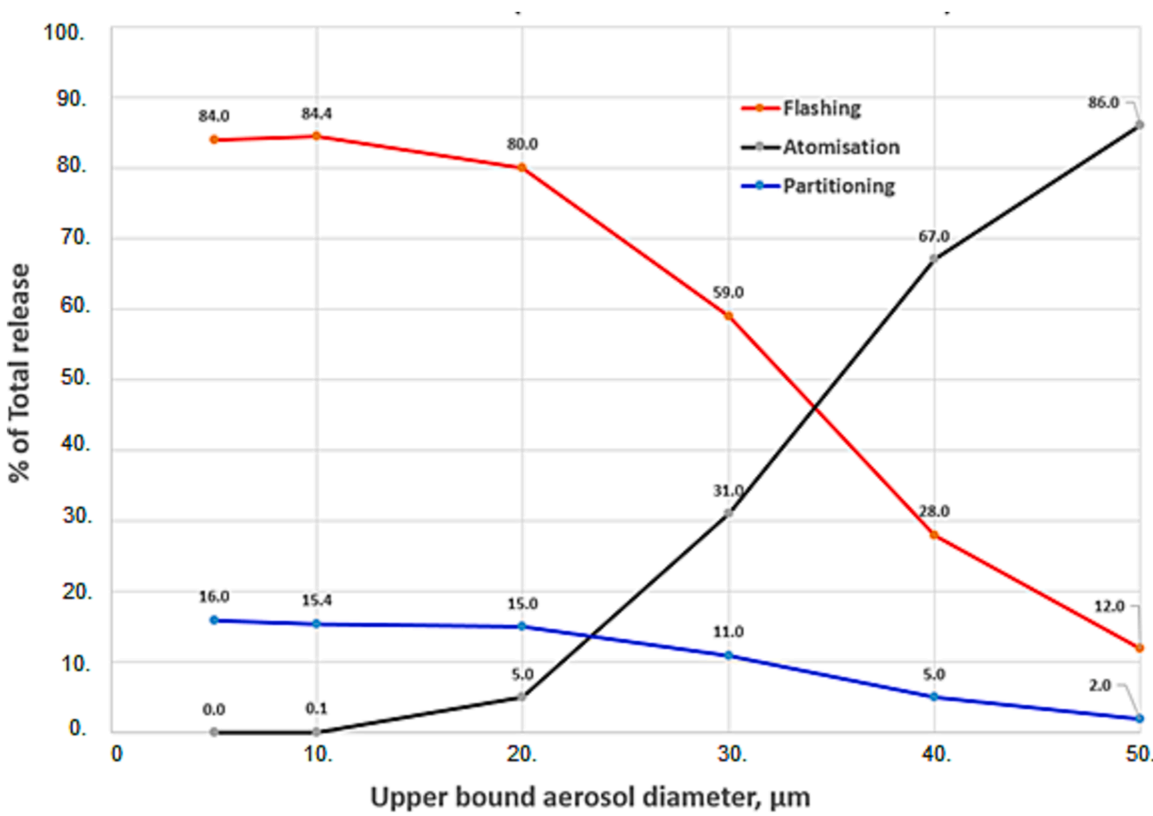

Fig. 12. Impact of the droplet sizes on the total released activity to the environment.

draft, Methodology, Formal analysis. **A. Malkhasyan:** Methodology, Formal analysis. **M. Salmaoui:** Methodology, Formal analysis. **L.E. Herranz:** .

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

No data was used for the research described in the article.

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Disclaimer

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