Nuclear Engineering and Design 417 (2024) 112877

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

Application of REPAS to analyze the sump clogging issue following a LOCA and its impact on the reliability of the ECCS long-term core cooling function

Andrea Bersano[a,*], Gianmarco Grippo[a], Giuseppe Agnello[b], Enrico Zio[c,d], Fulvio Mascari[a]

[a] FSN-SICNUC-SIN, ENEA, Italy
[b] Dipartimento di Ingegneria, Università di Palermo, Italy
[c] MINES Paris, PSL University, CRC, France
[d] Dipartimento di Energia, Politecnico di Milano, Italy

ARTICLE INFO

Keywords:
REPAS
Sump clogging
TRACE
Long-term core cooling

ABSTRACT

Sump clogging has been identified as a relevant issue after an accident occurred in the Barsebäck-2 nuclear power plant in Sweden (1992). Following a steam line Loss of Coolant Accident (LOCA) due to the inadvertent opening of a safety relief valve, the jet stripped some insulation material from nearby pipes. The insulation debris were transported to the inlet of the strainers for the drywell spray system, thus clogging the intake. The accident was not serious but showed that the Emergency Core Cooling System (ECCS) could have failed. As a consequence, several actions have been undertaken by international organizations, regulatory bodies and nuclear power plant owners to characterize this issue, and propose solutions and improvements. In the present paper, the Reliability Evaluation of Passive Safety Systems (REPAS) methodology is applied to analyze the sump clogging issue and its effect on the long-term core cooling function. Originally developed to evaluate the reliability of passive systems, REPAS is here applied for the first time to an active system. The application is performed using the TRAC/RELAP Advanced Computational Engine (TRACE) code, developed by the USNRC, to simulate a generic three-loop PWR, with an active decay heat removal system.

1. Introduction

Long-term core cooling is a fundamental function to be provided in several accident sequences concerning Nuclear Power Plants (NPPs). In the initial part of the transient it is usually performed through the Safety Injection System (SIS), until the depletion of the water inventory in the dedicated tanks. Then, long-term core cooling is obtained by recirculating the water collected in the sump below the reactor cavity. To avoid possible obstructions in the sump circuit, strainers are installed at the inlet, to block materials that may collect in the sump.

At the Barsebäck-2 NPP in Sweden, a shutdown BWR with external pumps and a containment of Mark-II design (OECD/NEA/CSNI, 1996), in 1992, following a Loss Of Coolant Accident (LOCA), the clogging of the sump strainers occurred, due to the accumulation of fibrous material stripped by the primary coolant jet from the pipe insulation. The accident, although not serious, showed that the active sump circulation could have failed, jeopardizing long-term cooling of the core (OECD/NEA/CSNI, 1994, Zigler et al., 1995, OECD/NEA/CSNI, 1996, OECD/NEA/CSNI, 2002, Hart, 2004, OECD/NEA/CSNI, 2013).

In the present paper, the reliability of the long-term cooling function by sump recirculation is evaluated applying the Reliability Evaluation of Passive Safety Systems (REPAS) methodology (D'Auria and Galassi, 2000, Ricotti et al., 2002, Bianchi et al., 2002, Jafari et al., 2003, Pierro et al., 2009, D'Auria, 2014). REPAS was developed by ENEA, University of Pisa, Polytechnic of Milan and University of Rome to evaluate the reliability of a passive system operating in natural circulation. In

Abbreviations: BAF, Bottom of Active Fuel; BEPU, Best Estimate Plus Uncertainty; DAKOTA, Design Analysis Kit for Optimization and Terascale Applications; ECCS, Emergency Core Cooling System; FC, Failure Criteria; HPIS, High Pressure Safety Injection System; LOCA, Loss of Coolant Accident; LPIS, Low Pressure Safety Injection System; NPP, Nuclear Power Plant; PCS, Primary Cooling System; PDF, Probability Density Function; PWR, Pressurized Water Reactor; REPAS, Reliability Evaluation of Passive Safety Systems; RPV, Reactor pressure Vessel; RWST, Refueling Water Storage Tank; SG, Steam Generator; SIS, Safety Injection System; SNAP, Symbolic Nuclear Analysis Package; SOT, Start Of the Transient; TAF, Top of Active Fuel; TM, Target Mission; TRACE, TRAC/RELAP Advanced Computational Engine.

* Corresponding author.
E-mail address: andrea.bersano@enea.it (A. Bersano).

https://doi.org/10.1016/j.nucengdes.2023.112877
Received 16 October 2023; Received in revised form 11 December 2023; Accepted 21 December 2023
Available online 29 December 2023

particular, in (D'Auria and Galassi, 2000) the REPAS methodology was applied to an isolation condenser, whereas in (Jafari et al., 2003) an application to the TTL-1 facility was presented to optimize the system design and estimate its thermal-hydraulic reliability.

The present work is the first in which the REPAS methodology is applied to an active system. The blockage of sump circulation due to the accumulation of material at the strainers may be considered analogously to the effect of instabilities that can disrupt natural circulation in a passive system. A preliminary analysis was presented in (Bersano et al., 2021).

The application has been carried out for a generic three-loop PWR-900 modeled with TRAC/RELAP Advanced Computational Engine (TRACE), a best-estimate thermal-hydraulic system code developed by USNRC. TRACE v5.0 patch 6 has been used (U.S. Nuclear Regulatory Commission, 2020) and the adopted nodalization has been developed through the Symbolic Nuclear Analysis Package (SNAP) (Applied Programming Technology, Inc., 2012). The selected transient is a double-ended Large Break LOCA on the Cold Leg (CL) of one of the three loops.

2. Description of REPAS

The REPAS methodology was developed for the evaluation of the reliability of passive systems, whose function is strictly connected to the driving thermal-hydraulic phenomena. Once the Target Mission (TM) of the system is identified (e.g. remove heat from the core), it is possible to define the (functional) Failure Criteria (FC) (e.g. energy removed lower than a certain threshold) and compute it by simulation with a reference computer code (e.g. TRACE in the present work) with a qualified nodalization.

The code calculation is deterministic whereas for the evaluation of the reliability of the system it is necessary to take into account the uncertainties in the phenomena involved and the associated parameters. To account for these uncertainties, the system design and critical parameters must be identified and the related ranges of variability and probability density functions defined. Then, repeated code runs can be performed, sampling the input design and critical parameters values[1] and computing the system functional response related to the previously identified FC, for estimating the reliability of the system.

To sum up, the REPAS methodology can be summarized in the following main steps (Fig. 1):

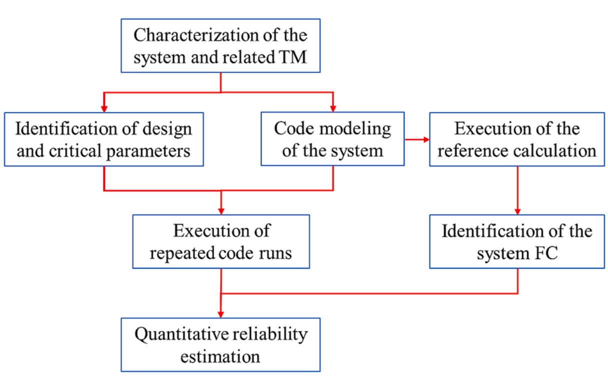

Fig. 1. Main steps of the REPAS methodology.

- Characterization of the design of the system, its operating conditions and TM.
- Identification of design and critical parameters, with their reference values, ranges of variability and probability distributions.
- Code modeling of the system and execution of the reference calculation.
- Identification of the system FC.
- Execution of repeated code runs.
- Quantitative reliability estimation.

Details on the REPAS methodology can be found in (D'Auria and Galassi, 2000, Ricotti et al., 2002, Bianchi et al., 2002, Jafari et al., 2003, Pierro et al., 2009, D'Auria, 2014).

3. Description of TRACE code and application

3.1. *TRACE code*

TRACE is a best-estimate thermal-hydraulic system code developed by the USNRC (U.S. Nuclear Regulatory Commission, 2020). It is a component-oriented code developed for best-estimate analysis of LWRs. In particular, TRACE was designed for the simulation of operational transients and LOCAs, and to model the thermal-hydraulic phenomena taking place in the experimental facilities used to study the steady-state and transient behavior of reactor fission systems (Mascari et al., 2011, Mascari et al., 2016). The code is based on two-fluid, two-phase field equations. This set of equations consists in the conservation laws of mass, momentum and energy for the liquid and gas fields (Mascari et al., 2012). A description of TRACE code can be found in (Mascari et al., 2016, U.S. Nuclear Regulatory Commission, 2020, Mascari et al., 2012).

3.2. *Generic three-loops PWR-900 nodalization*

The TRACE nodalization of the generic three-loops PWR-900 was originally developed in (Bersano and Mascari, 2019) and further refined in (Agnello et al., 2022). The model is composed of 78 Hydraulic Components and 49 Heat Structures. For the present analysis, the containment has been modified introducing an axial subdivision and the sump circulation system (pump, heat exchanger, piping and valves) has been added. Fig. 2 shows a sketch of the adopted nodalization.

The three loops of the reactor (Loop A, Loop B and Loop C) are modelled separately. The pressurizer (PRZ), located in the Loop B, is modelled using the "pressurizer" component available in TRACE. The break is located in the CL of the Loop B and it has been modelled with a set of three valves: at the Start Of the Transient (SOT), one valve interrupts the connection between the two lines of the CL and, simultaneously, the other two valves connect it to the containment.

In the PWR-900 TRACE nodalization, the implemented ECCS are:

- 3 hydraulic accumulators.
- 3 High Pressure Safety Injection Systems (HPIS).
- 3 Low Pressure Safety Injection System (LPIS).
- The sump circulation system.

The HPIS and LPIS are modeled together through a fill component and all the injections are performed in the CLs. A fixed mass is available for the HPIS and LPIS injections and when the value is reached the sump circulation begins.

The accumulators are modelled through three pipe components, which inject water into the Primary Cooling System (PCS) if the primary pressure falls below 40 bar.

Finally, the Reactor Pressure Vessel (RPV) is modeled through the 3D Vessel component available in TRACE with 2 radial sectors, the inner one for the core and the outer one for the downcomer, 3 azimuthal sectors, one for each primary loop, and 7 axial sectors (Bersano and Mascari, 2019). Both the nominal power and decay power are provided

[1] In general, different sampling methods can be adopted in the REPAS methodology. In a complete application, it should be demonstrated that the sampling method selected does not affect the results. In this regard, the "extreme cases" are helpful to support the probabilistic results. In the present application a Monte Carlo sampling has been adopted.

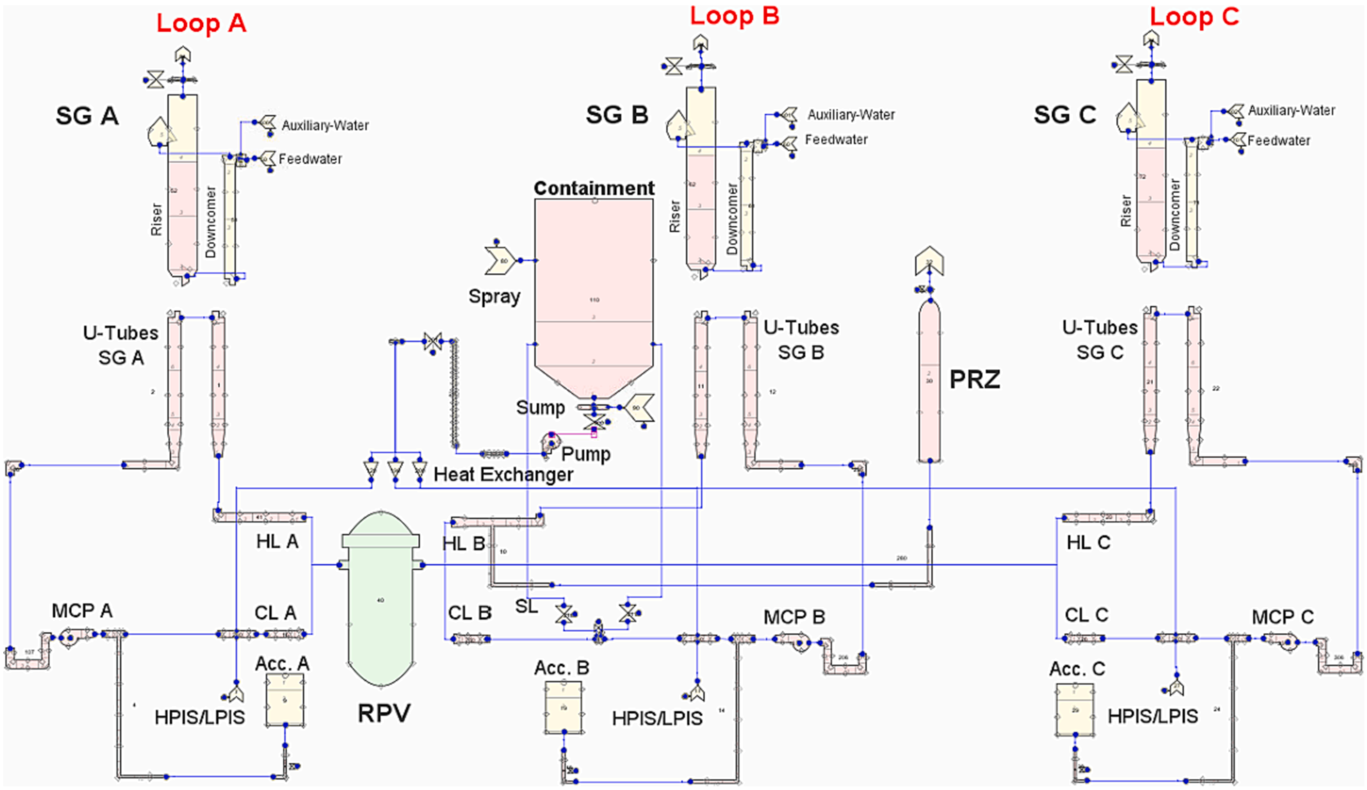

Fig. 2. TRACE nodalization developed by SNAP.

by three HSs through a single power component.

Considering the need for executing several code runs, the nodalization has been developed with a tradeoff between the level of detail and the required computational time.

3.3. Parameters for the REPAS application

Considering the adopted nodalization, the phenomena and quantities possibly affecting the unreliability of the selected safety system have been identified:

- Liquid mass trapped in containment compartments.
- *Heat losses from the containment.
- *Core power.
- Sump hydraulic diameter.
- Primary-to-containment heat transfer coefficient.
- *Strainer flow area.
- *Strainer form loss coefficient.
- Sump pump inlet line pressure losses (it could be important for pump cavitation).
- *Heat transfer in the heat exchanger of the decay heat removal system.
- Debris accumulation in the reactor primary coolant system (flow area reduction in the core lower plate).
- Sump pump elevation from sump bottom.
- *Sump pump characteristic.
- *Volume of safety injection system tanks.

Given the purpose of demonstrating the applicability of REPAS to an active system, only a subset of all parameters has been considered for the analysis, specifically the parameters marked with an "*" in the previous list. These have been chosen considering the features of the adopted nodalization.

4. Reference calculation results

The reference calculation is performed to set the FC. In the present reference calculation, the sump flow area fraction value is the maximum one (i.e. sump 100 %). A complete long-term core cooling analysis would require at least 72 h of simulation time. Since the main purpose of this analysis is to show the applicability of REPAS methodology to active systems, only 15000 s of simulation have been performed in total (1000 s of steady state and 14000 s of transient calculation). The required computational time is about 3 h.

4.1. Steady state results

In the present case 1000 s of steady state calculation are performed before the start of the transient. Table 1 shows some relevant steady state calculated parameters for the primary and the secondary systems, which have been compared against reference values from (Mascari et al., 2019).

4.2. Transient results

At the SOT (t = 0 s), due to the double-ended guillotine break on the cold leg of Loop B, the reactor SCRAM occurs and the transient

Table 1
Calculated steady state parameters.

Parameter	Value
Reactor power [MW]	2785
Primary cooling system pressure [bar]	156
Cold leg flow rate [kg/s]	4737
RPV flow rate [kg/s]	14,204
Inlet core temperature [K]	560
Outlet core temperature [K]	594.6
Secondary cooling system pressure [bar]	58
Feedwater flow rate SGs [kg/s]	512

progression is determined by the activation of the available ECCS. The main events are summarized in Table 2.

The reference transient progression is shown in Figs. 3-7, considering the primary pressure, the RPV mass flow rate, the hot rod maximum cladding temperature, the RPV collapsed coolant level and the containment pressure, respectively.

At the break opening, the primary pressure drops in the blowdown phase and the RPV mass flow rate reverses due to the break location in the cold leg. The core uncovering occurs very soon (Bottom of Active Fuel (BAF) uncovering at 6.4 s after the SOT) and the first cladding temperature peak is observed at 5.5 s.

The refill phase lasts for 14 s; the RPV collapsed coolant level reaches a minimum and, then, reaches again the BAF at 46 s. In the following reflood phase, the second cladding temperature peak occurs with a value of 781 K. Then, the core is fully rewetted and the RPV collapsed liquid level remains above the Top of Active Fuel (TAF) for the remaining part of the simulation.

Fig. 7 shows the containment pressure behavior along the transient. After the break opening, a peak close to 4 bar occurs; then, the pressure gradually decreases due to the containment spray activation. At the beginning of the sump circulation (around 3924 s), the pressure slightly increases reaching a final value of about 1.5 bar.

5. REPAS application

5.1. Failure criteria definition

For the present REPAS application, three FC have been identified based on the reference calculation results:

1) *Maximum cladding temperature*. In the reference calculation, the final cladding temperature is around 400 K (Fig. 5).

FC1: cladding temperature >600 K after the start of sump circulation;

2) *RPV collapsed level*. In the reference calculation, the core is always covered by the coolant after the end of the reflood phase (Fig. 6).

FC2: RPV collapsed coolant level <3/4 of the active core after the start of sump circulation;

3) *Containment pressure*. In the reference calculation, the final containment pressure is around 1.5 bar (Fig. 7).

FC3: Containment pressure >2.0 bar after the start of sump circulation.

It is important to underline that in the present work the FC definitions are not related to engineering or safety limits, rather they are defined based on the reference calculation, only to show an exemplificative application of the REPAS methodology.

Table 2
Sequence of events.

Event	Time after the SOT [s]
SOT	0
SCRAM	0.017
TAF uncovering	2.7
First cladding temperature peak	5.5
Injection accumulator loop B	3.5
BAF uncovering	6.4
Injection accumulator Loops A and C	9.5
Starting refill phase	32
Starting reflood phase	46
Second cladding temperature peak	46
End of reflood	145
Starting sump recirculation	3924

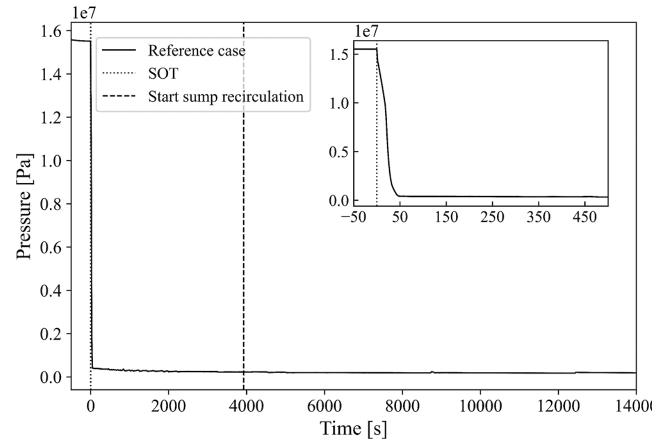

Fig. 3. Primary pressure.

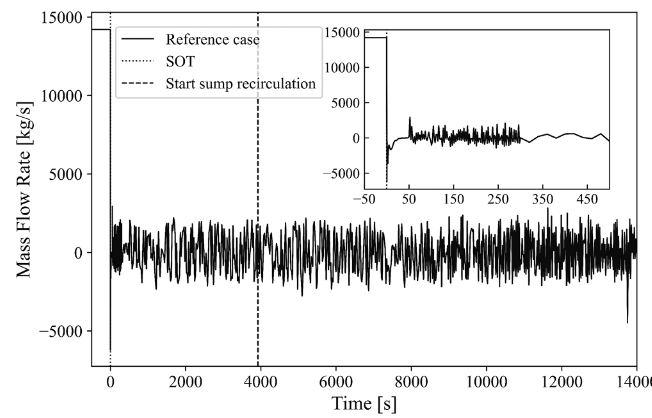

Fig. 4. RPV mass flow rate.

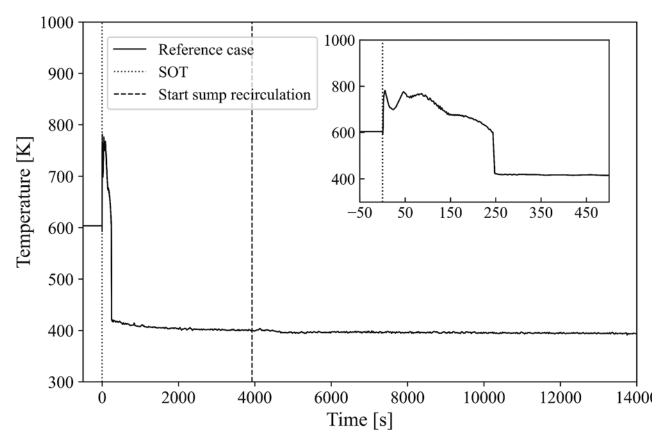

Fig. 5. Hot rod maximum cladding temperature.

5.2. Selection of calculations

The REPAS application has been conducted performing calculations with the TRACE code. The calculations have been set up both probabilistically, with the values of some input parameters selected by Monte Carlo sampling, and deterministically, with the values of some input parameters chosen by the authors, as done in (Jafari et al., 2003). The deterministic calculations have been included to consider somewhat limit cases of low probability of occurrence that are unlikely to be sampled in the probabilistic calculations. In particular, the deterministic

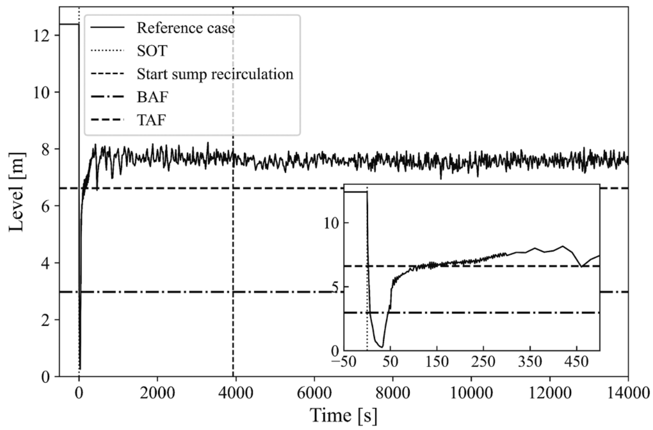

Fig. 6. RPV collapsed coolant level.

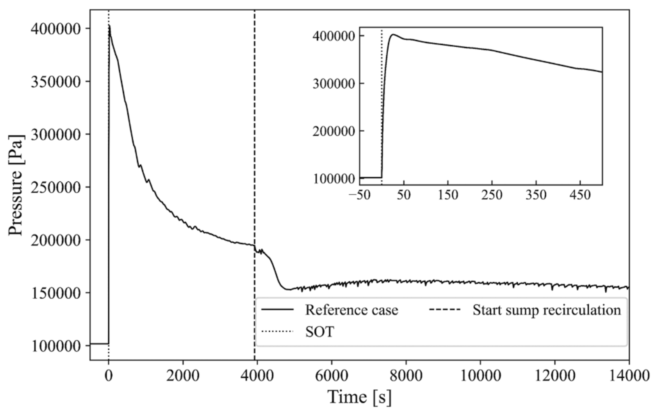

Fig. 7. Containment pressure.

calculations have been added to consider low values of sump opening area and high value of decay power. Deterministic calculations in the present analysis are also called "extreme cases", relative to the ranges of variation of the parameters. In this exemplificative application, no probabilistic analysis has been conducted to estimate the probability of occurrence of the selected deterministic cases, and probability values have been assumed only to show the application of the methodology.

Considering the probabilistic calculations, the required number of repeated calculations is defined in a way to obtain the convergence of the values of the figures of merit selected to demonstrate (in this case) the safety of the analyzed system. The authors' experience in the application of similar methodologies confirms that the number of calculations depends on the number of selected input parameters (differently e.g. from the Wilks method (Wilks, 1941; Wilks, 1942)), on the range of variation and on the complexity of the system. Furthermore, based on previous experiences in reliability evaluations and uncertainty quantifications, it is expected that the number of calculations is in the order of several hundreds. However, the current study considered has an exemplificative purpose and only a relatively small number of calculations can be carried out for the application of the presented methodology to the analysis of interest.

The probabilistic calculations have been set up using the Design Analysis Kit for Optimization and Terascale Applications (DAKOTA) uncertainty analysis plug-in available in SNAP (Applied programming Technology Inc., 2012a). For the probabilistic calculations, 200 runs have been executed. The parameters selected to set up the probabilistic calculations are reported in Table 3, each one with the corresponding Probability Density Function (PDF) selected. The input parameters are considered to be independent. As previously mentioned, only a subset of

Table 3
Selected parameters for the REPAS probabilistic code calculations.

Parameter	Reference value	PDF
Sump opening ratio	1	**Histogram** 1 %, $0 \leq x < 0.05$ 2 %, $0.05 \leq x < 0.1$ 17 %, $0.1 \leq x < 0.5$ 80 %, $x \geq 0.5$
Containment outer side heat transfer coefficient	10 W/m^{2}K	**Histogram** 15 %, 0 W/m^{2}K $\leq x <$ 5 W/m^{2}K 70 %, 5 W/m^{2}K $\leq x <$ 15 W/m^{2}K 15 %, 15 W/m^{2}K $\leq x \leq$ 20 W/m^{2}K
Core power scaling factor	1	**Normal** (mean 1, standard deviation 0.02)
Sump heat exchanger outer wall temperature	293.15 K	**Histogram** 17.5 %, 283.15 K $\leq x <$ 285.15 K 65 %, 285.15 K $\leq x <$ 301.15 K 17.5 %, 301.15 K $\leq x \leq$ 303.15 K
Sump strainers minor loss coefficient	100	**Histogram** 55 %, $100 \leq x < 150$ 25 %, $150 \leq x < 175$ 20 %, $175 \leq x \leq 200$
RWST mass offse	0 m^3	**Histogram** 17.5 %, −1.692E5 kg $\leq x <$ −8.4601E4 kg 65 %, −8.4601E4 kg $\leq x <$ 8.4601E4 kg 17.5 %, 8.4601E4 kg $\leq x \leq$ 1.692E5 kg
Pump sump volumetric flow rate	0.20 m^3/s	**Uniform** (min 0.18 m^3/s, max 0.22 m^3/s)

the identified parameters has been selected since the goal is to show an exemplificative application of the methodology. The core power scaling ratio PDF has been taken as in (Perez et al., 2011).

5.3. Results

The results of the probabilistic and deterministic calculations, with reference to the three FC identified, are presented in Fig. 8, Fig. 9 and Fig. 10, respectively. In addition, in the Figures it is marked the start time of the sump circulation as occurring in the reference calculation. Note that this time may vary in the probabilistic calculations due to the variation of the Refueling Water Storage Tank (RWST) mass (through the RWST mass offset parameters).

5.3.1. Probabilistic calculations

FC1 is not met by any probabilistic calculation (Fig. 8), since the maximum cladding temperature is always below 600 K after the start of the sump circulation.

FC2 is met by 7 out of 200 probabilistic calculations (Fig. 9). However, in those calculations the reduction of the RPV level below the FC2 threshold occurs only for limited periods of time due to local oscillations. Therefore, the core dry-out does not occur and the maximum cladding temperature (Fig. 8) remains far below 600 K (FC1 not met, as pointed out above).

Considering the third FC (containment pressure), it is met by 16 out of 200 probabilistic calculations (Fig. 10). 13 calculations cross the threshold just after the beginning of sump circulation due to a local pressure peak, whereas just 3 calculations are above the FC3 in the long term. It can be noted that in some cases the pressure final value may be even lower than the reference calculation, due to the selected

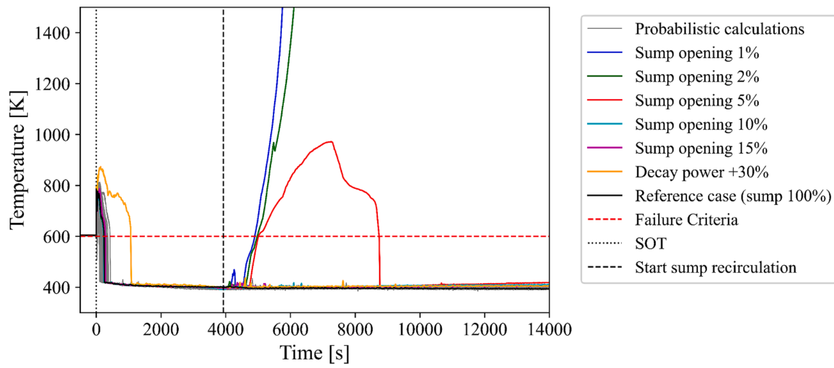

Fig. 8. Maximum cladding temperature.

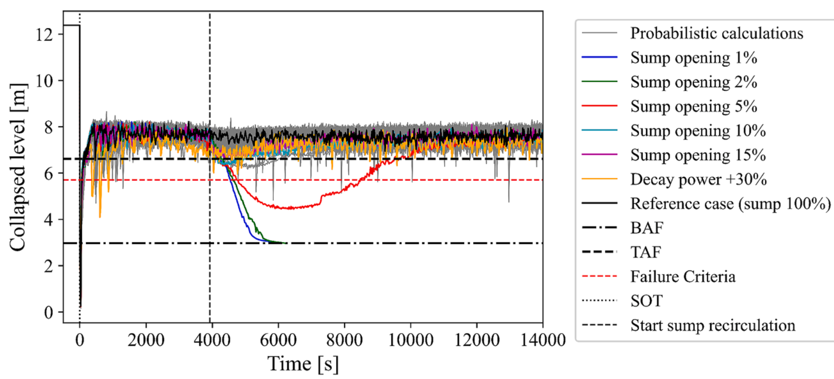

Fig. 9. RPV collapsed coolant level.

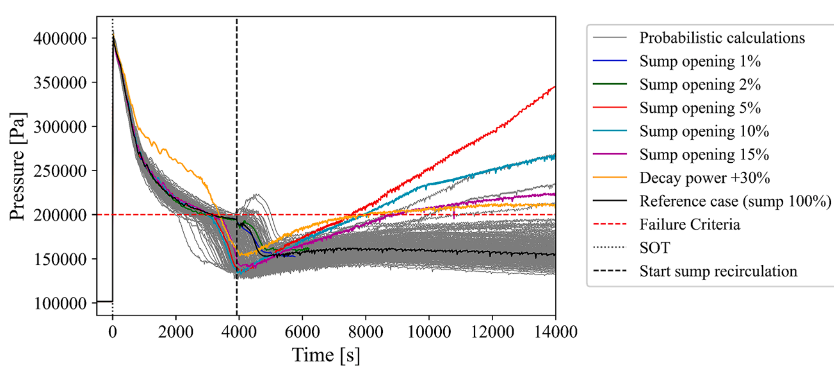

Fig. 10. Containment pressure.

parameters (e.g. higher containment outer heat transfer coefficient with respect to the reference calculation).

It should be reminded that only 15000 s have been simulated, which is a relatively small time for a long-term core cooling analysis. Considering a longer time might lead to some runs reaching the FCs due to the long-term effects of the selected parameters.

5.3.2. *Deterministic calculations*

Considering the results of the deterministic calculations, the first FC is met in three cases: sump opening ratio of 1 %, 2 % and 5 %. At sump opening ratio of 5 % the maximum cladding temperature overcomes the FC threshold for around 4000 s but, then, the core is completely quenched and the temperature returns to around 400 K. At sump opening ratio of 1 % and 2 % the complete core uncovering occurs and the temperature continues to rise up to the calculation stop.

Considering FC2, whose behavior is related to FC1, it is met by opening ratios of 1 % and 2 %, where a complete core uncovering occurs, and opening ratio of 5 %, where the collapsed level reduces below around half of the active core and, then, rises up above the TAF.

Concerning the last FC, the containment pressure is above the selected threshold in the cases with sump opening ratios of 5 %, 10 % and 15 %. FC3 seems not met with sump opening ratios of 1 % and 2 %. However, in these cases the simulation stops due to the reaching of the TRACE cladding temperature limit (melting temperature); therefore, it is expected that FC3 would be met also in these cases if the calculations had continued. In addition, FC3 is met also in the case of decay power increased by 30 %. In this case, the energy provided to the system is higher and the heat removal in the sump heat exchanger is not sufficient to prevent the pressure increase above the threshold. This case was added as an example of "extreme case" to evaluate the system behavior under very challenging conditions, despite the extremely low probability of occurrence.

5.3.3. System reliability estimation

The final step of the REPAS application consists in the estimation of the (functional) failure probability of the system. Once more it should be underlined that, as previously discussed, this is only an exemplificative application. Therefore, the probability value of failure here provided is not necessary realistic.

Fig. 11, Fig. 12 and Fig. 13 show respectively the maximum cladding temperature, the minimum RPV collapsed level and the maximum containment pressure after the start of sump circulation, as a function of their probability *p* of occurrence. The occurrence probability refers to the occurrence of the set of sampled (or imposed) input parameters' values for each simulation. This probability has been computed by the authors in a spreadsheet, considering the sampled values of the input parameters in each code run and their PDF. Based on these data, it has been computed the probability of occurrence of the set of input parameters in each code input-deck, which provides a certain output value. The FC are also highlighted in the graphs. The points with a high probability (in the order between 10^{-4} and 10^{-2}) are related to the probabilistic calculations, whereas the points with low probability are related to the deterministic ones. The results are in accordance with the findings of the previous sections:

- FC1 is met only by the 3 deterministic calculations with the lowest probability.
- FC2 is met by the 3 deterministic calculations with the lowest probability and 7 probabilistic calculations with a higher probability.
- FC3 is met by 16 probabilistic calculations and by all deterministic calculations (except the cases with the lowest probability due to the calculation stop).

Based on these results, the probabilities of occurrence of each single FC and of at least one of the three FCs have been computed, as reported in Table 4. Having the probability of occurrence of each code run, the probability of each FC has been computed considering the code runs where that specific FC is met. Finally, it has been computed the probability of having at least one FC considering the probability of all runs where at least one FC is met. As expected from the previous results, FC1 has the lowest probability of occurrence, whereas FC2 and FC3 have a similar high probability of occurrence since also some probabilistic calculations meet these FCs. The probability of occurrence of at least one FC has the same order of magnitude of FC2 and FC3, since they are larger than the probability of the FC1 by some orders of magnitude.

6. Conclusions

Sump clogging can be a relevant issue in long-term core cooling, as shown by the Barsebäck-2 NPP accident in 1992, because it can reduce the reliability of the sump circulation system.

In the present paper, the REPAS methodology, initially developed for the reliability evaluation of passive systems, has been applied to the sump circulation active system. The aim was not to carry out a full reliability evaluation of the system but, rather, to show the applicability of the methodology also to active systems.

A number of parameters affecting the system response have been selected and calculations have been performed with the best-estimate thermal-hydraulic system code TRACE in SNAP, considering a generic three loop PWR-900 reactor. The results of the application have shown:

- The applicability of the REPAS methodology to an active system. This implies that this method can be adopted to compare an active and a passive system with the same target mission.
- That deterministic calculations (or extreme cases) are helpful to analyze system states with low probability, which would then be unlikely to be selected in the probabilistic calculations.

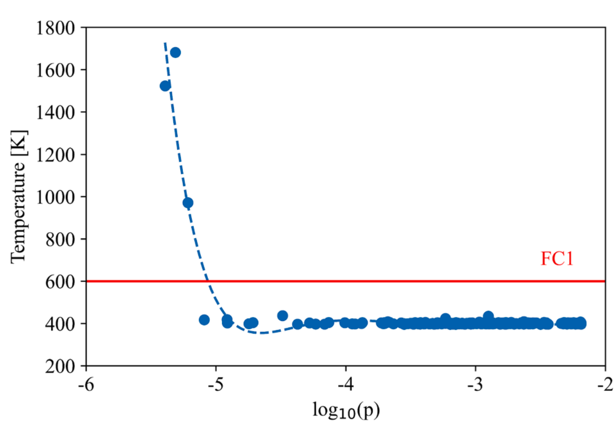

Fig. 11. Maximum cladding temperature as a function of the occurrence probability.

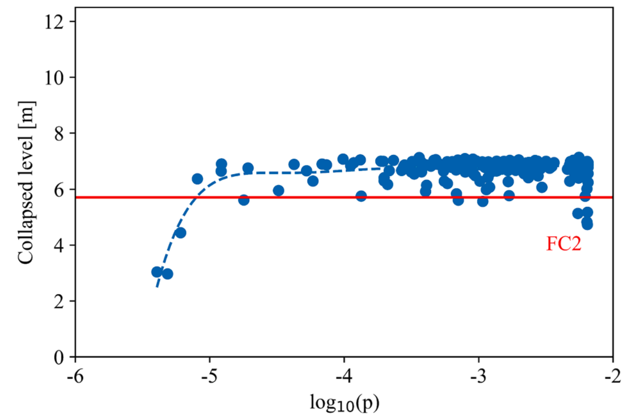

Fig. 12. Minimum RPV collapsed level as a function of the occurrence probability.

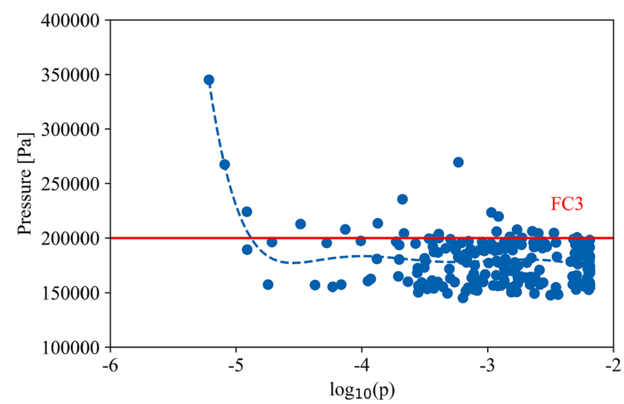

Fig. 13. Maximum containment pressure as a function of the occurrence probability.

Table 4
Probability of occurrence of FC1, FC2, FC3 and of at least one FC.

Event	Value
P(FC1)	1.501E-05
P(FC2)	2.646E-02
P(FC3)	2.353E-02
P(one FC)	4.892E-02

- That in the adopted nodalization, the clogging of the sump area leads to meeting the failure criteria only for very low opening ratio values (<10 % for FC1);
- That in the present case, FC2 (RPV level) and FC3 (containment pressure) are the failure criteria with the highest probability of occurrence. However, it should be reminded that in this analysis the FCs are not related to engineering or safety limits, rather they are defined based on the reference calculation, only to show an exemplificative application of the REPAS methodology.

Finally, it should be underlined the relation between REPAS and the Best Estimate Plus Uncertainty (BEPU) approach. REPAS is a methodology to evaluate the reliability of a system, and therefore its unreliability region. BEPU is an approach to apply a deterministic code to a system with the quantification of the results uncertainties. Therefore, even if in the REPAS application some aspects may seem similar to an uncertainty analysis (e.g. sampling of the input parameters, use of tools like DAKOTA, etc.), their meaning is different. However, in a complete reliability evaluation performed adopting system codes, e.g. applying REPAS methodology, the code uncertainties should be taken into account in a BEPU approach. Therefore, according to the authors opinion, REPAS should be applied in a BEPU framework and this may modify the unreliability region computed by the code, with respect to the actual unreliability region of the system.

CRediT authorship contribution statement

Andrea Bersano: Conceptualization, Methodology, Formal analysis, Investigation, Writing – original draft. **Gianmarco Grippo:** Formal analysis, Investigation. **Giuseppe Agnello:** Formal analysis, Investigation. **Enrico Zio:** Conceptualization, Methodology, Writing – review & editing. **Fulvio Mascari:** Conceptualization, Methodology, Resources, Supervision, Writing – review & editing.

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 authors are unable or have chosen not to specify which data has been used.

Acknowledgements

The authors gratefully wish to acknowledge the inspiration and support of Prof. Francesco D'Auria in the development of the activity described in this paper.

References

Agnello, G., Di Maio, P.A., Bersano, A., Mascari, F., 2022. Cold Leg LBLOCA uncertainty analysis using TRACE/DAKOTA coupling. Journal of Physics: Conference Series 2177, 012023.

Applied programming Technology Inc., 2012a, "Uncertainty analyses User manual, Symbolic Nuclear Analyses Package (SNAP)".

Applied Programming Technology, Inc., 2012, "Symbolic Nuclear Analysis Package (SNAP) User's Manual".

A. Bersano, G. Agnello, F. D'Auria, E. Zio, F. Mascari, 2021, "Methodology and Application to Characterize the Sump Clogging Issue in case of Long-Term Core Cooling", RCCS-2021-OECD/NEA Specialist Workshop on Reactor core and containment cooling systems – long term management and reliability.

Bersano, A., Mascari, F., 2019. Evaluation of a Double-Ended Guillotine LBLOCA Transient in a Generic Three-Loops PWR-900 with TRACE Code Coupled with DAKOTA Uncertainty Analysis. *ATW-International Journal for Nuclear Power* Issue 11/12, 526–532.

F. Bianchi, L. Burgazzi, F. D'Auria, M.E. Ricotti, 2002, "The REPAS approach to the evaluation of passive safety systems reliability", Proceedings Of an international workshop on passive system reliability – A challenge to reliability engineering and licensing of advanced nuclear power plants, NEA/CSNI/R(2002)10, Cadarache, France.

D'Auria, F., Galassi, G.M., 2000. Methodology for the Evaluation of the Reliability of Passive Systems. DIMNP NT 420 (00).

F. D'Auria, 2014, "Reliability of TH Passive Safety Systems: Background", IAEA Training Course on Natural Circulation Phenomena and Passive Safety Systems in Advanced Water-Cooled Reactors, Mumbai, India.

G.H. Hart, 2004, "A Short History of the Sump Clogging Issue and Analysis of the Problem", *Nuclear News*, pp. 24-34, March 2004.

Jafari, J., D'Auria, F., Kazeminejad, H., Davilu, H., 2003. Reliability evaluation of a natural circulation system. Nuclear Engineering and Design 224, 79–140.

F. Mascari, G. Vella, B. G. Woods, K. Welter, F. D'Auria, 2012, "Analysis of Primary/Containment Coupling Phenomena Characterizing the MASLWR Design During a SBLOCA Scenario", *Nuclear Power Plants*, InTech.

F. Mascari, F. De Rosa, B.G. Woods, K. Welter, G. Vella, F. D'Auria, 2016, "Analysis of the OSU-MASLWR 001 and 002 Tests by Using the TRACE Code", NUREG/IA-0466.

F. Mascari, J. C. De La Rosa Blul, M. Sangiorgi, G. Bandini, 2019, "Analyses of an Unmitigated Station Blackout Transient in a Generic PWR-900 with ASTEC, MAAP and MELCOR Codes", NUREG/IA-0515.

Mascari, F., Vella, G., Woods, B.G., Welter, K., Pottorf, J., Young, E., Adorni, M., D'Auria, F., 2011. Sensitivity analysis of the MASLWR helical coil steam generator using TRACE. Nuclear Engineering and Design 241, 1137–1144.

OECD/NEA/CSNI, 1994, "Proceedings of the OECD/NEA Workshop on the Barseback Strainer Incident", NEA/CSNI/R(1994)14, Stockholm, Sweden.

OECD/NEA/CSNI, 1996, "Knowledge base for emergency core cooling system recirculation reliability", NEA/CSNI/R(95)11.

OECD/NEA/CSNI, 2002, "Knowledge Base for Strainer Clogging – Modification performed in different countries since 1992", NEA/CSNI/R(2002)6.

OECD/NEA/CSNI, 2013, "Updated Knowledge Base for Long Term Core Cooling Reliability", NEA/CSNI/R(2013)12.

Perez, M., Reventos, F., Batet, L., Guba, A., Tóth, I., Mieusset, T., Bazin, P., de Crécy, A., Borisov, S., Skorek, T., Glaeser, H., Joucla, J., Probst, P., Ui, A., Chung, B.D., Oh, D. Y., Pernica, R., Kyncl, M., Macek, J., Manera, A., Freixa, J., Petruzzi, A., D'Auria, F., Del Nevo, A., 2011. Uncertainty and sensitivity analysis of a LBLOCA in a PWR Nuclear Power Plant: Results of the Phase V of the BEMUSE programme. Nuclear Engineering and Design 241, 4206–4222.

F. Pierro, D. Araneo, G. Galassi, F. D'Auria, 2009, "Application of REPAS Methodology to Assess the Reliability of Passive Safety Systems", *Science and Technology of Nuclear Installations*, Article ID 768947.

M.E. Ricotti, F. Bianchi, L. Burgazzi, F. D'Auria, G. Galassi, 2002, "The REPAS Study: Reliability Evaluation of Passive Safety Systems", Proceedings of The 10th International Conference on Nuclear Engineering (ICONE10), pp. 245-254, Arlington, USA.

U.S. Nuclear Regulatory Commission, 2020, "TRACE v5.0 patch 6 Theory Manual".

Wilks, S.S., 1941. Determination of sample sizes for setting tolerance limits. Annals of Mathematical Statistic 12 (1), 91–96.

Wilks, S.S., 1942. Statistical prediction with special reference to the problem of tolerance limits. Annals of Mathematical Statistic 13 (4), 400–409.

G. Zigler, J. Brideau, D.V. Rao, C. Shaffer, F. Souto, W. Thomas, 1995, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris", NUREG/CR-6224.