BEPU analysis of a 2-in DVI break in a generic IRIS SMR by ASTEC – RAVEN coupling

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Abstract:
Advanced light-water Small Modular Reactors (SMRs) have acquired great interest in the international framework due to recognized advantages in terms of flexibility, capital cost and especially safety features, achieved by their designs. The “inherent safety” of SMRs is guaranteed by lower decay-heat, allowing the use of integral configuration for the primary coolant system and of passive safety systems. In the framework of NUGENIA TA-2 ASCOM collaborative project, coordinated by IRSN, an ASTEC code generic input-deck, based on an IRIS-like reactor, has been recently developed with the aim of studying the code capability to simulate SMR designs in challenging conditions. In the present paper, a Best-Estimate Plus Uncertainty (BEPU) method has been applied to study the safety criteria in a Design Basis Accident (DBA) due to a 2-in guillotine break of a Direct Vessel Injection (DVI) line and by assuming the operability of the passive safety systems of the generic IRIS. The uncertainty quantification study is performed through the propagation of the input uncertainty methodology, by implementing the RAVEN-ASTEC coupling on a multi-core cluster. The input uncertainty parameters perturbing the system are selected among the main reactor’s initial and boundary conditions as well as related to passive safety systems. The statistical study of the reactor response in terms of output variation of the main safety Figures Of Merit (FOMs) is carried out by analyzing the sensitivity of the FOMs, with respect to the variation of the input uncertainties. The study is aimed to provide information regarding the role played by passive safety systems in the mitigation strategy; to characterize the thermal-hydraulic response of the code model and its capability to simulate the main natural-driven phenomena of passive advanced SMRs; and to develop a first uncertainty analysis regarding ASTEC application to SMR.

1 INTRODUCTION

The growing global energy demand and the need for sustainable and carbon-free energy production entail great progress to be achieved in relation to the energy mix. In this framework, advanced SMRs are considered as a key design options for the development of nuclear technology, because of their “inherent safety” due to a lower nominal power, adoption of passive safety systems and integral configuration. The ASTEC code [1] (Accident Source Term Evaluation Code), developed by the French “Institut de Radioprotection et de Sûreté Nucléaire” (IRSN), aims at simulating an entire Severe Accident (SA) sequence in nuclear water-cooled reactors from the initiating event through the release of radioactive elements. In the framework of NUGENIA TA-2 ASCOM project [2], an ASTEC code input-deck of a generic IRIS reactor [3] has been recently developed with the aim of studying the code capability to simulate SMRs in accidental conditions. The present paper proposes a Best Estimate Plus Uncertainty (BEPU) methodology to study the uncertainty affecting the ASTEC simulation of a DBA sequence in the generic IRIS model. The input sources of uncertainty for the simulation have been selected among those related to the operation of passive safety systems; the output FOMs are chosen to fit the most relevant safety conditions of the reactor. The preliminary Uncertainty Quantification (UQ) study is aimed at providing some insights on the behavior of the FOMs and characterizing the sensitivity of some relevant uncertain input parameters. In adddition, the study provides valuable information regarding the ASTEC simulation thermal-hydraulics and characterizing the role played by the passive safety systems in the mitigation strategy.

2 DESCRIPTION OF ASTEC CODE AND GENERIC IRIS MODEL

2.1 ASTEC severe accident code

The ASTEC code [1] features a modular structure where each module is aimed at simulating a set of physical phenomena or related to a reactor zone. In the present work, the modules CESAR, CPA and ICARE have been used for the simulation of the investigated DBA. CESAR [1] is dedicated to the simulation of coolant systems thermal-hydraulics, it is a two-phase system code based on a two-fluid type model. CPA [1] is aimed at simulating the thermal-hydraulics taking place in the reactor containment. ICARE [1] is dedicated to core internals heat-exchange and in-vessel degradation. The ASTEC version used is the V2.2.0.

2.2 Description of IRIS reactor and ASTEC modelling

IRIS is an integral PWR of 300 MWe. Its integral Reactor Pressure Vessel (RPV) hosts all the components of the primary coolant system (core, pressurizer (PRZ), Steam Generators (SGs), Reactor Coolant Pumps (RCP), etc.) avoiding large vessel penetration. A spherical steel containment encloses the RPV and most of the passive safety systems and it is directly involved in the mitigation strategy. The passive safety systems of IRIS include an Emergency Heat Removal System (EHRS), using the Refueling Water Storage Tanks (RWST) as heat sink; two stages of Automatic Depressurization System (ADS); Emergency Boration Tanks (EBTs); Long-term Gravity Make-up System (LGMS); Pressure Suppression System (PSS). EBT and LGMS can inject cooling water in the vessel through DVI lines. Fig 1 summarizes the IRIS design; more details can be found in [3][4].

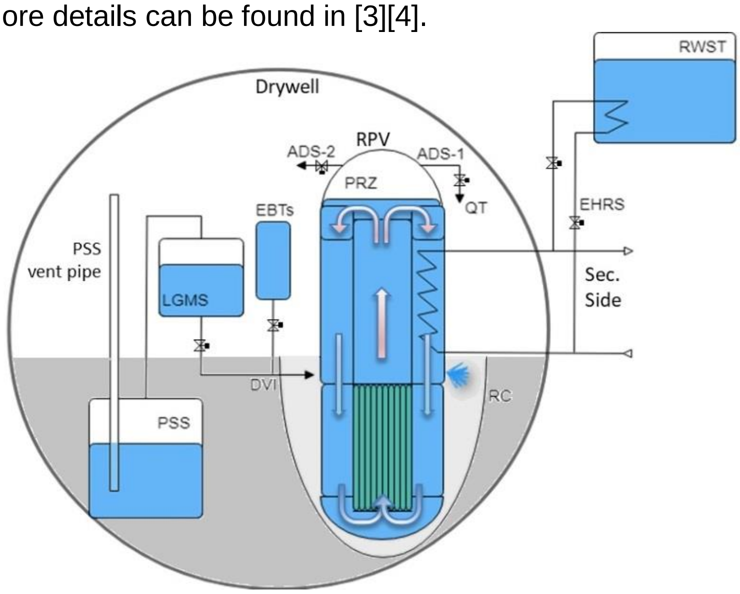

Fig. 1: Scheme of the main components and passive safety systems for a generic IRIS reactor [4].

The approach followed in the generic IRIS modelling with ASTEC V2.2.0 aims at accurately simulating the thermal-hydraulics of RPV and passive safety systems. For this reason, CESAR (Fig. 2-left) has been used for the nodalization of the primary systems (top half of RPV); secondary system (feed and steam lines; SGs) and most of the passive safety systems. The lower half of RPV, including the core, has been modelled with ICARE (Fig. 2-

right). CPA completes the nodalization modeling the spherical containment (Drywell (DW) and Reactor Cavity (RC)). More details on the nodalization can be found in [4].

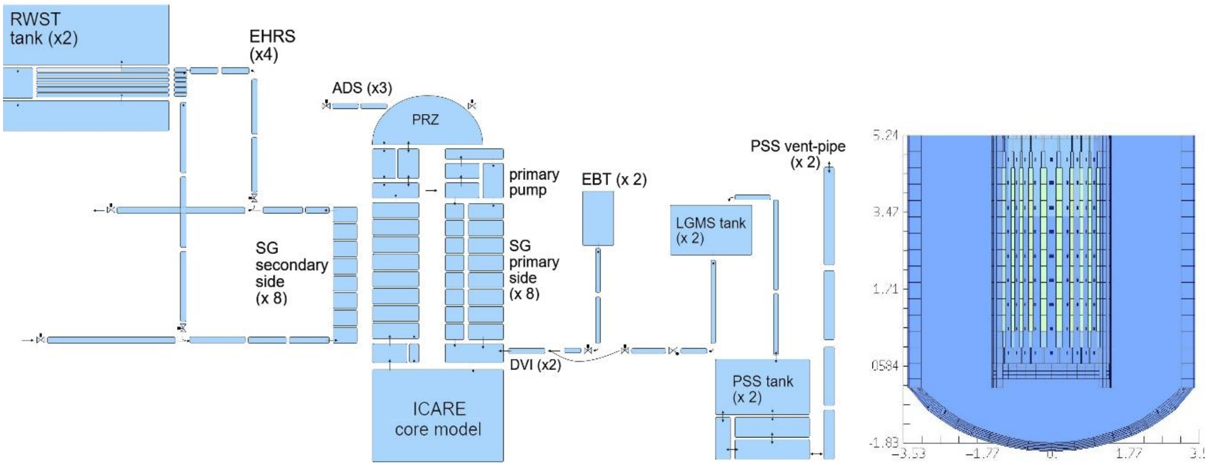

Fig. 2: Schemes of CESAR (left) and ICARE (right) nodalization of the generic IRIS reactor [4].

3 ASTEC SIMULATION OF DBA SEQUENCE

Among the DBA scenarios studied in the past for IRIS, the guillotine break of one DVI line is the most challenging accident in terms of safety [4]. The ASTEC simulation of the sequence has been performed after 2000 s of steady-state needed to reach the reactor nominal conditions. The transient is initiated (t=0 s) at the opening of the break. The calculation has been carried out for 70000 s and Table 1 reports the main timings characterizing the simulation results.

Table 1: ASTEC simulation of DBA, main timings and systems actuations.

Event	Signal	Time	Actuation
DVI line-A break opening	-	0 s	-
High Containment P set-point	S-Signal	31 s	SCRAM; SS isolation; EHRS
Low PRZ level set-point	Low-PRZ	110 s	RCP coast-down, RI-DC valves
Low PRZ pressure set-point	LM-Signal	149 s	ADS stage-1, EBTs
EBT - A / B emptying	-	355 / 4100 s	-
Low DP (RPV-Cont.) set-point	Low DP	1335 s	LGMS
DW-PSS pressure inversion	-	1480 s	-
RC level at DVI level	-	4080 s	-
Low LGMS mass set-point	Low LGMS	19000 s	ADS stage-2
LGMS - A / B emptying	-	23000 / 27300 s	-
RWST water boiling	-	45100 s	-

The sequence is summarized by the following six phases : 1) The DVI break is followed by a rapid containment pressurization and RPV depressurization. The pressures of PSS and LGMS increase following the DW pressure. 2) At 31 s, the high DW pressure set-point triggers the SCRAM, secondary system isolation and EHRS actuation. Power starts to be removed from RPV by EHRS through natural circulation. 3) At 110 s, the low PRZ level triggers the RCP coast-down, and natural circulation starts in the RPV. 4) The low PRZ pressure set-point (149 s) triggers the ADS stage-1 opening, increasing the RPV depressurization. The valves of EBTs are opened and its injection takes place in the RPV and in the RC (intact and broken DVIs). 5) RPV - DW low pressure difference is reached at 1335 s and LGMS is actuated. The DW depressurization starts and its pressure decreases below PSS value. As a consequence, the code captures the PSS vent-pipes water level rising up, until dropping in the containment. 6) ADS stage-2 valves are opened at 19000 s, and RPV gets the same pressure of DW. During long-term cooling, the core is kept covered

thanks to RC water above the break. The residual heat is balanced by the power removal of EHRS and the heat losses from DW to environment.
From the analysis [4], not shown here for the sake of brevity, it can be concluded that the code is able to predict the main thermal-hydraulic phenomena driving the IRIS passive mitigation of the investigated DBA sequence.

4 UNCERTAINTY QUANTIFICATION ANALYSIS

4.1 Set-up of the UQ analysis

Among the UQ methodologies, the *probabilistic propagation of input uncertainties* is particularly suitable for code applications [5]. This method is based on the selection of Input Uncertain Parameters (IUPs), characterized by a Probability Density Function (PDF) and a range of variation. A set of N different simulations of the same sequence is run by varying the values of all the IUPs with a sampling strategy (e.g. Monte Carlo). The sampling size N can be selected by using the Wilks formula on the base of the chosen *confidence level* β, *probability content* γ and number of FOMs [5][6]. The UQ study has been carried out by coupling ASTEC with RAVEN [7], thanks to a Python interface. RAVEN manages the sampling and the simulations that are executed on a cluster; then it collects the ASTEC results and manages the statistical post-processing. The 7 IUPs selected for the study are summarized in Table 2. The choice aims at considering the main uncertainty affecting the operation of passive safety systems. PDF and range have been derived from previous studies [8][9] or by experts judgment by following a conservative approach. In order to ensure γ, $\beta \geq 95\%$, according to the method given in [6], for 2 FOMs and 1-side limit, at least 93 calculations are required. The chosen safety output FOMs are the cladding maximum temperature and the DW pressure. Both the FOMs have a maximum safety limit, respectively of 1204 °C (1477.15 K) and 13.5 bar.

Table 2: list of input uncertain parameters, range of variation and PDF type for UQ analysis.

Parameter	Name	Range	PDF	Reference
Power of decay heat	FpPow	[± 8 %]	normal	[8]
Friction form loss in PRZ surge line	KPrz	[0.5 – 2.]	normal	[8]
Heat-transfer surface of EHRS–RWST exchanger	SEhrs	[± 25 %]	uniform	[9]
RWST initial temperature & Env. temperature	TEnv	[10 – 30 °C]	uniform	Expert Judg.
Friction coeff. in subcritical condition of break	KfBrk	[± 30 %]	uniform	Expert Judg.
Friction form loss coefficient at DVI outlet	KDvi	[± 100 %]	uniform	Expert Judg.
Initial water level in LGMS tanks	LLgms	[± 10 cm]	uniform	[8]

4.2 Result of UQ analysis of the ASTEC simulation

All the calculations have been successfully completed. The DW pressure uncertainty has been described in terms of uncertainty band in Fig. 3-left (reference and maximum calculated values of the FOM are also reported in blue and in red, respectively). The safety condition has been satisfied all along the sequence and the maximum value of pressure registered is of 11.7 bar, at around 1050 s. The maximum band width of 1.2 bar, is reached at this time. The same plot has been reported in Fig. 3-right for the maximum cladding temperature. In this case also the safety criteria are always satisfied. Differently from the reference calculation, some of the UQ calculations features local temperature peaks around 1250 s, and the uncertainty band shows the maximum width of 28 K at this point. From a preliminary investigation, the peaks seem to be due to instabilities in the RPV natural circulation leading to a decrease of core level. More investigations on this point must be conducted in the future.

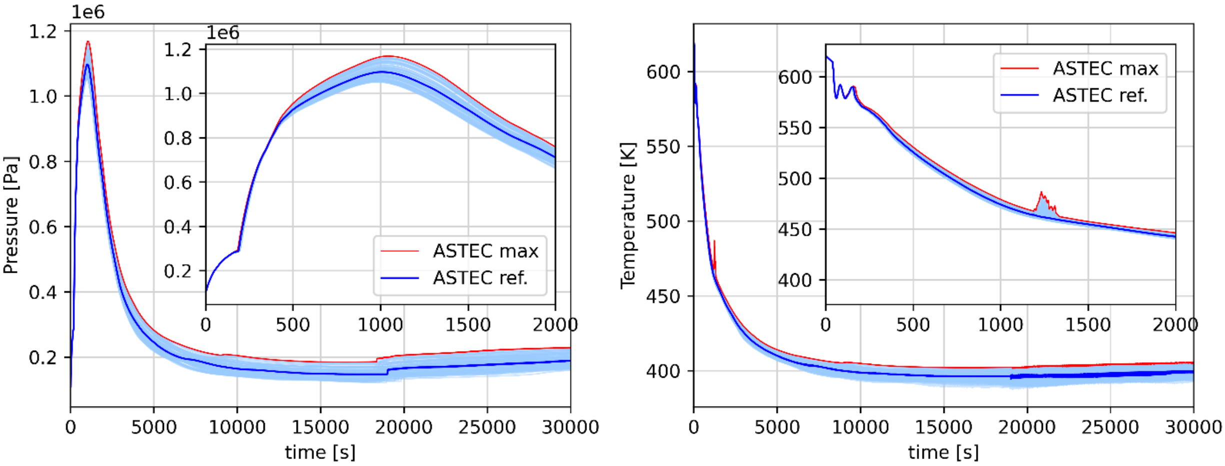

Fig. 3: Uncertainty band of the DW pressure (left) and of maximun cladding temperature (right).

4.3 Results of sensitivity analysis

The sensitivity of the two FOMs with respect to each IUP has been evaluated by means of sensitivity coefficients: the Spearman and Pearson correlation coefficients express the sensitivity in terms of monotonic and linear correlation respectively [5]; the Lasso coefficients are solution of the minimization problem of a Lasso regression [10]. For absolute value higher than 0.5, the sensitivity is classified as *significant*; between 0.2 and 0.5 it is assumed *moderate*; and below 0.2 it is *low* or *absent* [5]. Fig. 4 reports Spearman and Lasso coefficients related to DW pressure against the maximum calculated pressure. Pearson coefficients have been omitted since in all cases their values are very close to Spearman. It can be observed that at the highest pressure values and larger uncertainty band width (1050 s), representing the most challenging time with respect to this FOM, the main source of uncertainty is SEhrs parameter (heat-transfer surface of EHRS-RWST exchanger). A moderate positive sensitivity is also captured with TEnv and FpPow at this point.

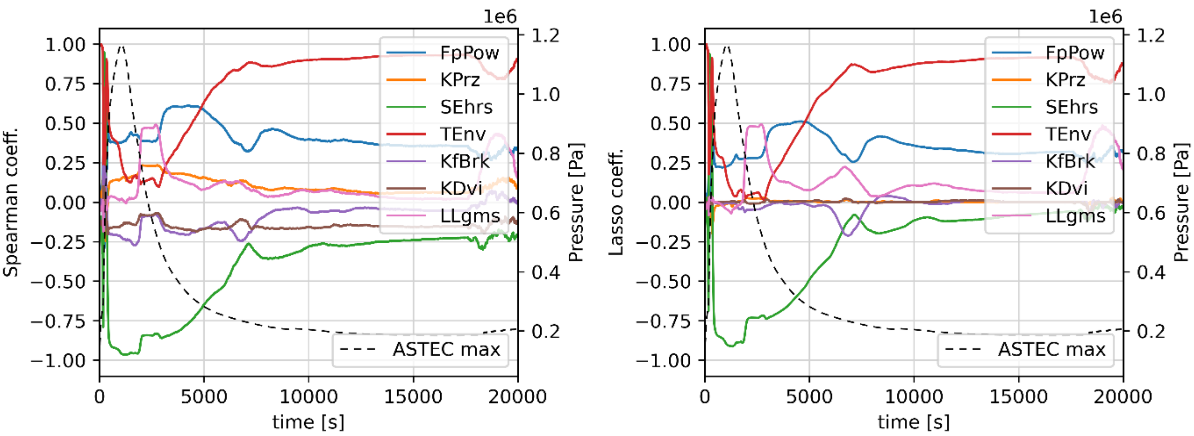

Fig. 4: Spearman (left) and Lasso (right) coefficients for DW pressure.

The same plots are reported in Fig. 5 for the cladding maximum temperature. As for the pressure, the main parameters showing sensitivity with the FOM are SEhrs and FpPow; TEnv features significant sensitivity at the onset and at the end of the sequence. It is important to underline that at the local temperature peaks (at about 1250 s) there is a local change in the sensitivity predicted: the sensitivity of SEhrs drops to low values, while there is a local increasing in the sensitivity of the FpPower and also of KPrz that reaches moderate values.

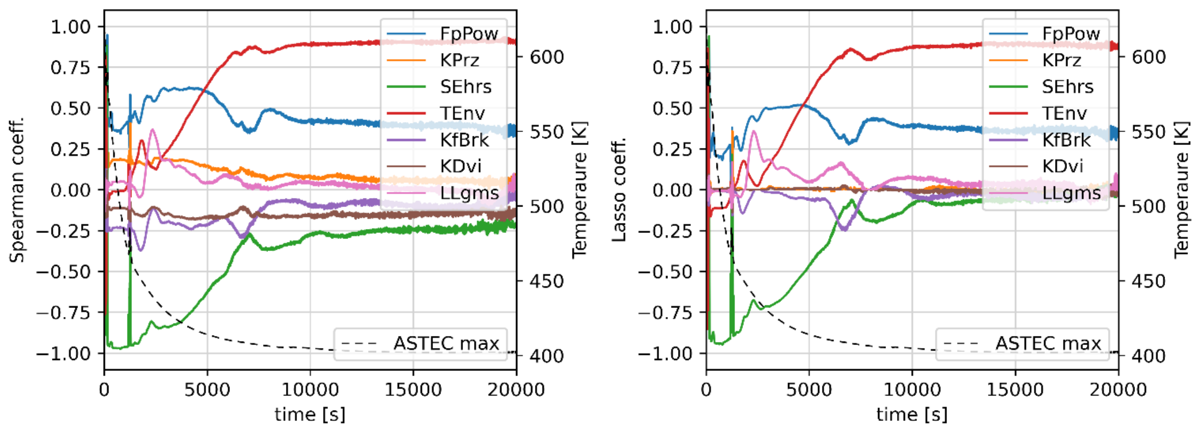

Fig. 5: Spearman (left) and Lasso (right) coefficients for maximum cladding temperature.

5 CONCLUSIONS

The paper proposes a BEPU analysis of a DBA transient in a generic IRIS reactor. The SMR model and the simulation of the passive mitigation strategy have been developed with the SA code ASTEC, V2.2.0. The UQ has been carried out to assess the propagation of input uncertainty by 7 selected IUPs on two FOMs. All the UQ calculations feature the same phenomenological evolution, and the two FOMs safety criteria are always satisfied along the sequence. The highest value registered for the DW pressure matches the maximum spread of uncertainty of this FOM at 1050 s and, at this point, the sensitivity coefficients show that the major source of uncertainty is the heat-transfer surface of EHRS-RWST exchanger. Besides suggesting an improvement in the knowledge of this IUP, this result underlines the key role played by EHRS system in the limitation of the DW pressurization. Moderate correlation to this FOM is also observed in the decay heat power and environment temperature parameters. Regarding the maximum cladding temperature, it features a quasi-monotonic decreasing behavior, with the exception of local peaks at around 1250 s showing up only in some of the UQ calculations. The maximum uncertainty band width of this FOM is registered at this point, with main contribution from the decay heat power and pressurizer surge line friction parameters. The analysis of natural convection instabilities observed at this point in the RPV should be the starting point for further studies. The significant sensitivity of the environment temperature parameter also highlights that in passively cooled reactors, simple but important precautions, as limiting the RWST temperature or the external temperature to DW surface, would results in an increase of the inherent safety of the plant.

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