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Preliminary Sensitivity Analysis for an Ex-Vessel LOCA without Plasma Shutdown for the EU DEMO WCLL Blanket Concept

Matteo D'Onorio[a], Fabio Giannetti[a], Maria Teresa Porfiri[b], Gianfranco Caruso[a]

[a] *Sapienza University of Rome, Dep. of Astronautical Electrical and Energy Engineering (DIAEE), Rome, Italy*
[b] *UTFUS-TECN, ENEA CR Frascati, Frascati, Italy*

In this early development phase of the DEMO design the uncertainty affecting many operational and design parameters can modify main outcomes of accident scenario aiming at studying the critical conditions for the vacuum vessel and the contiguous containment volumes. The aim of this paper is to perform a preliminary sensitivity analysis of an accident progression predicted by MELCOR code considering selected parameters as a figure of merit to predict possible code outcomes. The uncertainty band will be evaluated through sensitivity analyses programmed, collected and statistically manipulated through RAVEN software tool. MELCOR and RAVEN have been internally coupled through a new Python code interface developed by Sapienza University of Rome, to perform sensitivity and uncertainty quantification analyses during severe accident transient. The Beyond Design Basis Accident (BDBA) analysis of an ex-vessel loss of coolant accident (LOCA) for the water-cooled lithium lead (WCLL) blanket concept has been simulated with the fusion version of MELCOR 1.8.6 code. The postulated initiating event (PIE) is a double-ended break in the first wall (FW) cooling system distributor ring, with simultaneous failure of the plasma shutdown system. An in-vessel breach of the coolant system occurs because of FW failure, with consequent unmitigated plasma shutdown transient. Sensitivity analysis results have shown that the FW temperature at which plasma in-vessel breach occurs is strongly correlated with the mass of hydrogen produced. The same parameter has also an impact on the overall accident scenario, such as the trigger of VVPSS rupture disks and thus source term mobilization.

Keywords: *Sensitivity, RAVEN, MELCOR, DEMO, BDBA, Safety*

1. Introduction

As for fission power plants, the main environmental and safety issue for the future EU-DEMO fusion reactor [1] is the confinement of radioactive products into the reactor buildings during both normal operation and accidental conditions. For WCLL BB concept, the possibility of a tungsten-steam reaction during severe accidents is a safety concern because the hydrogen produced from the reaction could pose a flammability or detonation hazard [2]. Moreover, an over pressurization of either VV or TCR should be avoided during LOCA for both maintenance and safety reasons.

One of the potentially dangerous sources of hydrogen is related to the oxidation reactions of plasma-facing components (PFCs), such as first wall and divertor [3]. Preliminary safety studies conducted in the framework of the EUROfusion safety and environment (SAE) Work Package highlighted as BDBA accident event sequences, characterized by the failure of active plasma shutdown, can result in production of significant amounts of hydrogen because of the high temperature experienced by plasma-facing structures [4].

Safety studies commonly aim at obtaining an optimal nuclear power plant design and reactor operation, for this reason excess of conservatism should be avoided from the safety analyses. To obtain an in-depth safety assessment of the DEMO reactor the deterministic analyses should be combined with analyses implemented through a BEPU approach. In the past few years, an increasing interest in computational reactor safety analysis is to replace the conservative evaluation model calculations by best estimate calculations supplemented by uncertainty analysis of the code results. The sensitivity analysis, presented in this paper, has been performed through the RAVEN software tool by varying design and operational parameters which can affect the thermal behavior of FW structure and the pressure transient inside the VV. However, the scope of the analysis is not only to determine outcome variable values but more in general, to evaluate the overall system response for different combinations of the stochastic parameters.

2. RAVEN-MELCOR coupling

RAVEN (Risk Analysis Virtual Environment) [5,6] is a software tool, developed at the Idaho National Laboratory (INL) to act as a control logic driver and post-processing tool for different applications. Nowadays RAVEN is a multi-purpose probabilistic and uncertainty quantification platform, capable to be coupled with any system code. The software tool can be employed for several types of applications, such as uncertainty quantification [7], sensitivity analysis and probabilistic risk assessment [8]. A new Python code interface has been developed by Sapienza University of Rome to couple the MELCOR code with the RAVEN tool. The interface has three main functions: interpret the information coming from RAVEN; translate such information in the input of the driven code; manipulate output data file to create a database. To allow RAVEN storing output data coming from MELCOR, a Python output parser has been developed to convert the plot binary file generated by MELCOR into a CSV file. To overcome the handling of large datafiles the interface allows to create a CSV file with only variables required by the users. So, it is possible to obtain a database that comprises the required variables from all MELCOR

author's email: matteo.donorio@uniroma1.it

packages. Figure 1 shows the procedural framework used for sensitivity and uncertainty analyses. A MELCOR input deck is used as template, the chosen parameters are specified as string with special characters. RAVEN can identify such parameters and replaces the string with values sampled from a specified distribution. The sampled values are implemented into n number and consequently n-MELCOR input decks are generated. Data resulting from simulations are stored into a database that can be used to perform statistical analyses.

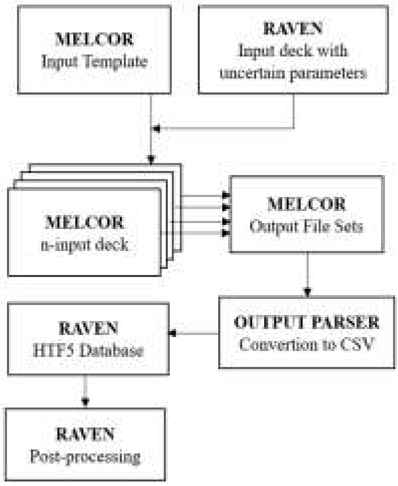

Fig. 1 RAVEN-MELCOR flow diagram

3. EU DEMO WCLL DESIGN

The EU-DEMO concept adopted as reference for this LOCA analysis is based on WCLL breeder blanket (BB) divided in 16 Sectors in toroidal direction [10]. The total fusion power produced is 1923 MW_{th}. The WCLL breeding basic unit is divided in the middle by a baffle plate for the PbLi part: the liquid metal enters to the cell bottom and then flows in radial-poloidal direction. The exit from the elementary cell top. Water flows in 20 U-shaped double-wall tubes (DWTs). The tubes are grouped and joined to the manifolds of the breeder zone (BZ) cooling water.

The FW is constituted by a 25 mm thickness of EUROFER with 2 mm tungsten wrap in the plasma facing area. FW (integrated in the module) has a square 7x7 mm cooling channels. Thermo-dynamic cycle of both circuits is based on a typical PWR conditions (295–328°C at 15.5 MPa). The total in-VV volume available for steam expansion is about 6400 m^3. The vacuum vessel pressure suppression system (VVPSS) is designed to mitigate the pressure in case of LOCA accidents. The actuation logic is described in [11].

4. Accident description

The PIE is a double-ended pipe break of the FW cold loop distributor ring inside the tokamak cooling room (TCR). Since there is no direct inherent feedback between transients in the cooling system and the plasma, an active system is required to terminate the plasma burn to limit the heat up of in-vessel components and to prevent ex-vessel events from propagating into an in-vessel loss of coolant event with the potential for significant hydrogen generation.

The failure of the plasma shutdown system has been assumed as an aggravating event after the PIE. Therefore, fusion power is not terminated. The uncooled FW modules are heated by nominal heat-flux coming from the plasma, leading to the failure of the EUROFER structure back to the FW tungsten layer.

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The reactor shutdown follows the injection of water in the plasma chamber and is characterized by an unmitigated plasma disruption that produces an additional in-vessel failure of the modules. The FW break has been simulated by opening a connection between the VV and the OB4 volume shown in figure 5.

After the two FW breaks, the coolant starts to leave the FW- primary heat transfer systems (PHTS) to enter the VV. Because the FW surfaces have been heated to high temperatures by radiation from the plasma, the steam ingress inside the vessel can result in production of hydrogen via W-steam reaction. Moreover, if upstream isolation valves are not installed or if their closure is slow, a bypass between the VV and TCR would allow for the possibility of mixing of hydrogen and air contained in the TCR vault.

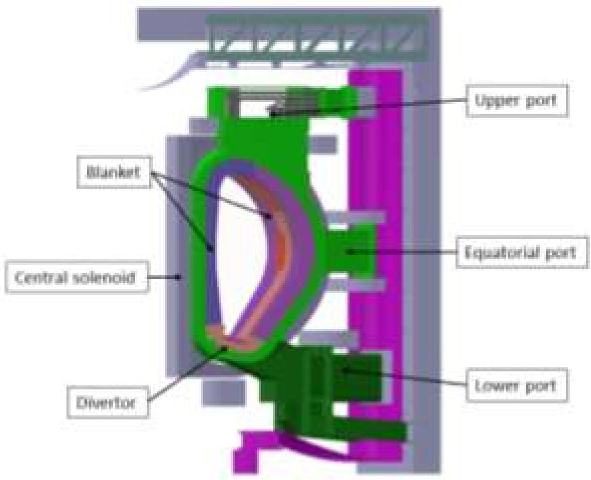

Fig. 2 DEMO baseline CAD in-vessel components

5. DEMO Modeling approach

The system code MELCOR 1.8.6 for fusion was used for this study [9]. The nodalization was developed to realize the sensitivity analysis using a 144-core cluster having a reasonable computational time, whilst maintaining realistic prediction of the phenomena involved during the accident. BZ and FW PHTSs [11] are modelled, with the addition of the VV and VVPSS to analyze the behavior of the accident.

5.1 PHTS nodalization

The whole EU-DEMO BB has been modelled, as in [12], with the division in three different regions simulating respectively 1 sector, a group of 7 sectors (from sector 2 to sector 8) and a group of 8 sectors (from sector 9 to sector 16). This kind of nodalization is realized to try the local "detailed" analysis for the single sector, considering the division of the PHTs in two parallel loops. Each sector has been modeled in order to investigate both inboard and outboard segments behavior during the accident sequence. The single segment concept is adopted, derived from WCLL BB 2017 designed. The segments have been divided into seven different outboard (OB) and inboard (IB) volumes (OB1 to OB7 and IB1 to IB7) to consider the poloidal differences in terms of free volumes, mass and heat.

5.2 VV and VVPSS nodalization

VV and VVPSS nodalization scheme is shown in Fig.4. The VV has been modeled with five control volumes to consider the right position of all the connections simulating the plasma chamber, the upper port, the lower port, the interspace volume between the divertor and the VV structure (CV852) and the interspace volume between the back-supporting structure (BSS) of BB modules and VV structure (CV853). The dimensions and free volumes of these zones have been evaluated from EU-DEMO baseline CAD model reported in fig.2, which represents the main in-vessel components. Flow area data are provided in Table 1. The developed nodalization for the VVPSS consists of 17 control volumes and 18 flow paths. Each tank has been modeled separately (CV906 - CV911) with the associated rupture disks (RD) and bleed lines (BL). BLs are triggered by the defined VV pressure limit of 90 kPa, while RDs act when VV pressure is 150 kPa. VVPSS components have been initialized at 40°C and 9.5 kPa.

5.3 Containment building nodalization

In Fig.4 the nodalization scheme of the EU DEMO WCLL containment is shown. The containment has been modeled with five control volumes (CV894 – CV898) to simulate the expansion volumes of the DEMO containment which CAD model is reported in fig. 3. An opening (FL972) between the FW-PHTS (FW cold ring in fig. 5) and the upper pipe chase volume has been provided to simulate the ex-vessel LOCA PIE. The total TCR volume available for steam expansion is 68300 m^3, because the top maintenance hall is not considered available for steam expansion.

Table 1. VV path flow area

FL	Flow Area [m^2]
FL974	0.577
FL975	2.352
FL976	1.790
FL977	6.300
FL978	0.107
FL979	6.330

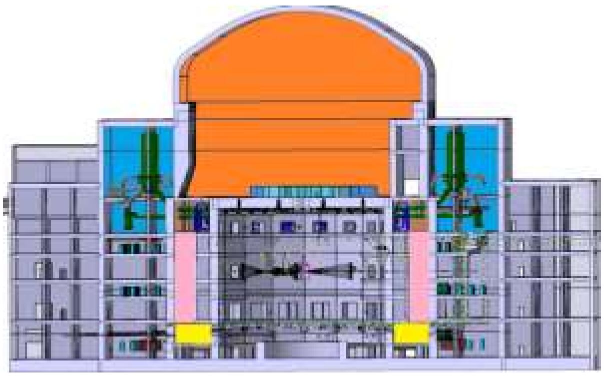

Fig. 3 EU-DEMO containment building

5.4 Heat structures

One-dimensional heat structures (HS) are modeled for the components such as the BB, the divertor and the VV. The FW is modeled into two heat structures. The first heat structure models the heat transfer between the plasma volume and the coolant in the FW. The second heat structure models the heat transfer between the FW coolant and the BZ, which is simulated with a multiple layer HS of EUROFER and LiPb. The total FW surface is equal to about 1445 m^2, 507.18 m^2 for the IB sectors and 937.79 m^2 for the OB sectors. A 130.2 m^2 two-layer tungsten-EUROFER HS is used to model the divertor system. The tungsten layer (12 mm thick) simulates the divertor plasma-facing components, while the

Tab.2 List of perturbed variables in MELCOR input-deck

Variable	Description	Mean μ	Sigma σ
FLARA_PIE	Distributor ring break area	0.04921 m^2	0.004921 m^2
FLARA_PD	FW break area after plasma unmitigated disruption	0.02568 m^2	0.002568 m^2
DISR_POW	Power deposited on FW by the plasma disruption	10 MJ/m²	0.5 MJ/m²
MOD_DH	Modules decay heat multiplicative factor	1.0	0.05
DIV_DH	Divertor decay heat multiplicative factor	1.0	0.05
ISL_CLOSE_P	Trip valve closure setpoint	12.965E+6 Pa	12.965E+5 Pa
T_BREAK	FW break temperature	1450.15 K	145.0
DIL_974	FL974 flow area	0.577 m^2	0.0577 m^2
DIL_975	FL975 flow area	2.352 m^2	0.0235 m^2
DIL_978	FL978 flow area	0.107 m^2	0.0107 m^2

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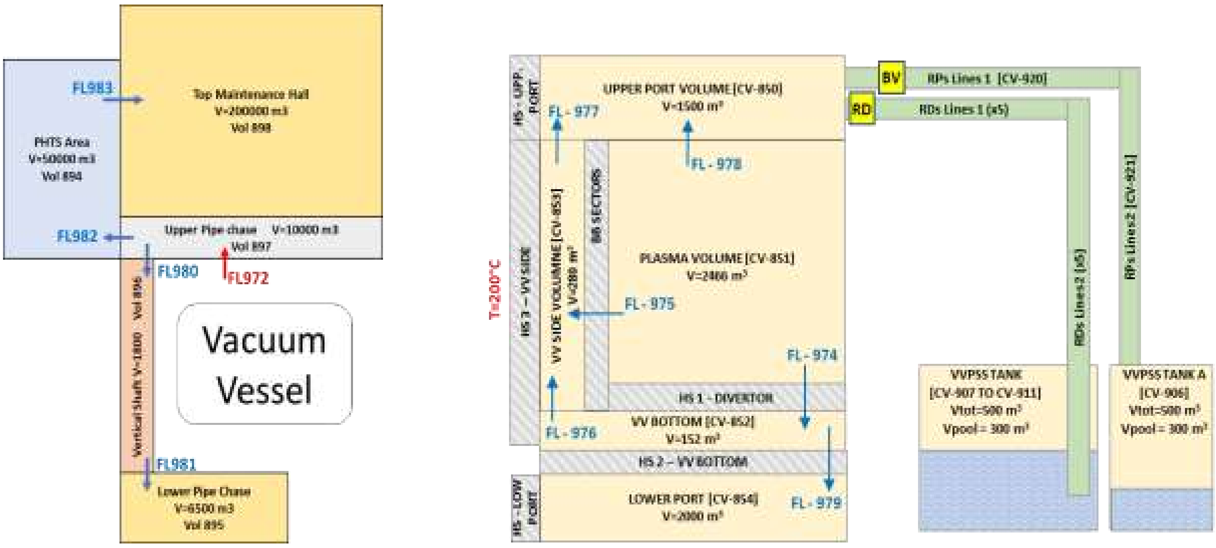

Fig. 4 DEMO TCR nodalization scheme / Vacuum Vessel and VVPSS nodalization scheme

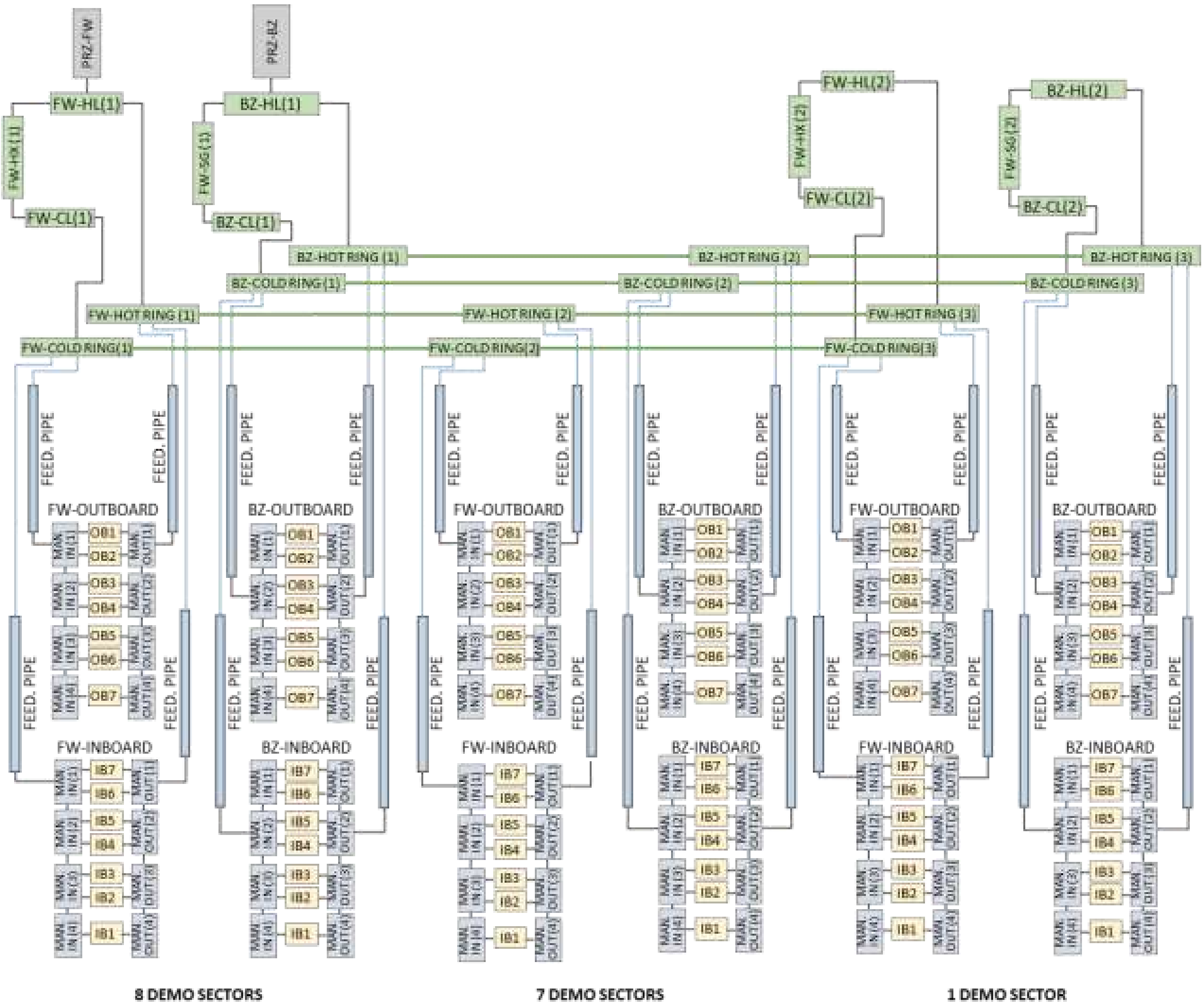

Fig. 5 - Thermal hydraulic MELCOR nodalization scheme adoped for DEMO FW and BZ PHTS

EUROFER layer (278.7 mm thick) simulates the divertor cassettes body.

6. Variables selection and sampling strategy

A set of 10 parameters has been chosen to perform this sensitivity study. The parameters are listed in table 2, together with the associated distribution. All the selected perturbations are based on a normal distribution. Upper and lower bound have been imposed in order to consider physical limits of perturbed parameters. The sample size needed to obtain a significative output statistic was selected using the Wilks formula for two-sided statistical tolerance limits [13]. The required minimum number of computer code calculations becomes 93 for a 95% probability and 95% confidence level [14].

A Monte Carlo sampling strategy has been used to randomly perturb the input space in relation to variable distributions, setting a limit of 300 calculations. The Monte Carlo method is one of the most-used sampling methodologies because does not use a structured discretization of the input space but cumulative probability and probability distribution function to compute the value to assign to a variable. The parameter to be perturbed have been selected as they may affect FW temperature transient and the steam partial pressure inside the VV.

7. Main outcomes from sensitivity analysis

A 2 hours transient after the postulated initiating event is simulated. The results from the MELCOR calculation are reported below. The transient starts after a 3000 s steady-state calculation. The accident transient starts at time t=0.0 s with the break of the distributor ring, opening a connection between the FW-PHTS and the TCR.

The maximum pressure reached in the containment is $1.4589 \cdot 10^5$ Pa, with a standard deviation of 2.815 kPa. Fig. 6 shows a scatter plot for the maximum pressure reached in the TCR and the flow area of the break in the FW distributor ring, with the related Pearson's coefficient. The general trend is so that as the distributor ring break area increases, the maximum pressure reached inside the TCR also increases. The time at which the maximum pressure peak inside the TCR is reached can ranges between 37.0 s and 106.5 s, after the FW-PHTS distributor ring failure.

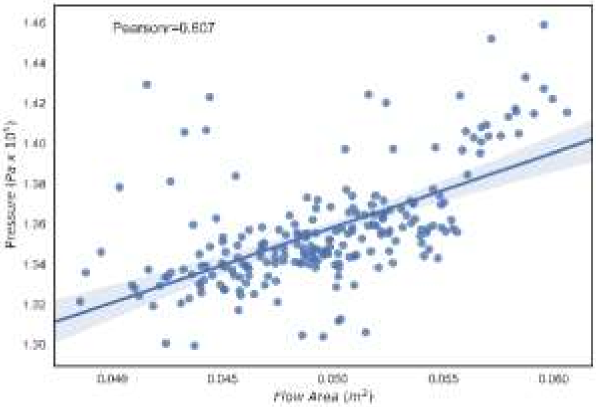

Fig. 6 Maximum pressure in the TCR vs break flow area

Fig. 7 shows the pressure evolution inside the VV for all the histories simulated. It is possible to notice how the perturbed variables can affect the pressure transient within the VV. Two main different trends can be distinguished. In fact, a high FW failure temperature causes a time delay in the in-vessel breach. Moreover, because of the long time required to close upstream trip valves (10 s after signal detection) large amounts of water are already released in the TCR. Therefore, when the in-vessel breach occurs the FW-PHTS pressure and inventory are such as to cause a slow pressurization in the VV, not enough to cause the opening of VVPSS rupture discs. The RDs, which allow for the discharging of steam and hydrogen in the VVPSS suppression tanks, are not triggered in the 70% of the simulated scenarios.

Results show that for FW failure temperature higher than 1450 K the VVPSS RDs are never triggered. In all these cases, hydrogen and other source term masses are not discharged inside the VVPSS-STs and can accumulate inside the TCR volumes. To avoid these worst accident scenarios, the pressure setpoint for the RDs opening should be decreased, at least below the 5% quantile reported in table 3. In such a way also these low-pressure in-vessel LOCA scenarios could be safely accommodated avoiding radioactive release inside the containment building.

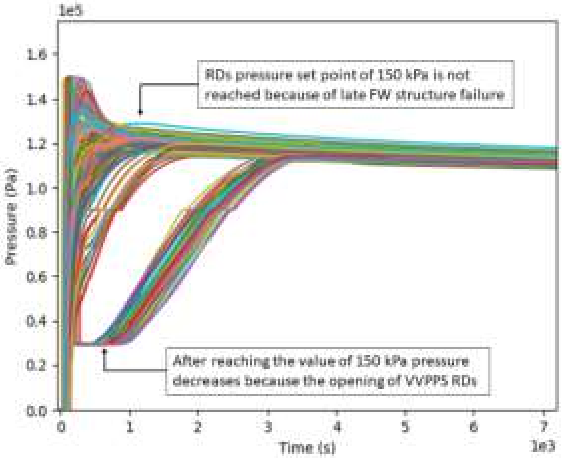

Fig. 7 VV pressure transient for each MELCOR run

Table 3. VV maximum pressure, descriptive statistic

Mean μ	Sigma σ	0.05 quantile	0.95 quantile
134.0 kPa	12.9 kPa	118.59 kPa	150.02 kPa

In fig.8 the correlation between the mass of hydrogen produced at the end of the simulation and the temperature characterizing the failure of the FW is shown, along with the univariate distribution of both variables on separate axes.

Hydrogen production is lower than 100 g when the FW temperature at which plasma in-vessel breach occurs is lower than 1590 K. For higher FW failure temperature, the total mass of hydrogen produced increases very quickly. However, the total mass never exceeds 800 g. It should be noted that the reaction between steam and tungsten dust deposited on the FW surface and on the divertor surface has not been considered in this simulation, because of the lack of an accurate model.

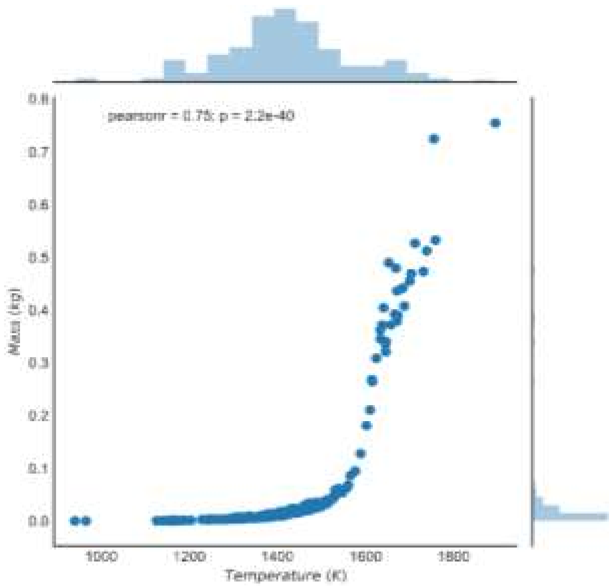

Fig. 8 Mass of hydrogen produced vs FW failure temperature

In table 4 the results of the sensitivity analysis for the hydrogen mass generated during the in-vessel phase of the accident progression are reported.

Table 4. Hydrogen mass production, descriptive statistic

Mean μ	Sigma σ	0.05 quantile	0.95 quantile
69.2 g	142.3 g	1.3 g	336.3 g

Hydrogen generation commences almost simultaneous with the failure of the FW armor and terminates and ends maximum 300 s after the PIE. The sudden injection of water steam inside the VV cools the plasma-facing structures (fig. 9) and ultimately reduces hydrogen production.

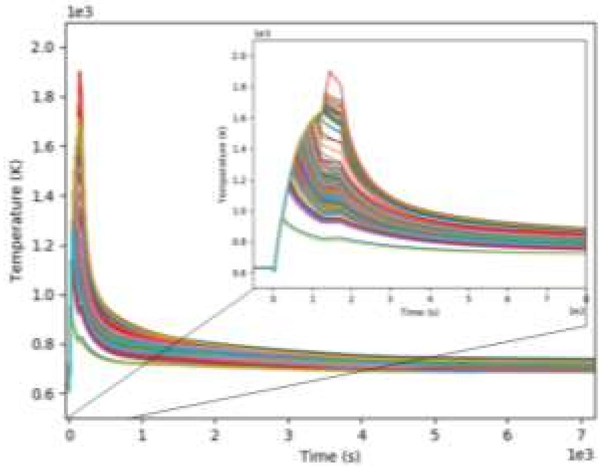

Fig. 9 FW temperature transient for each MELCOR run

Additional cooling to modules and VV is provided by the VV decay heat removal system which is considered in operation for the entire duration of the accident transient.

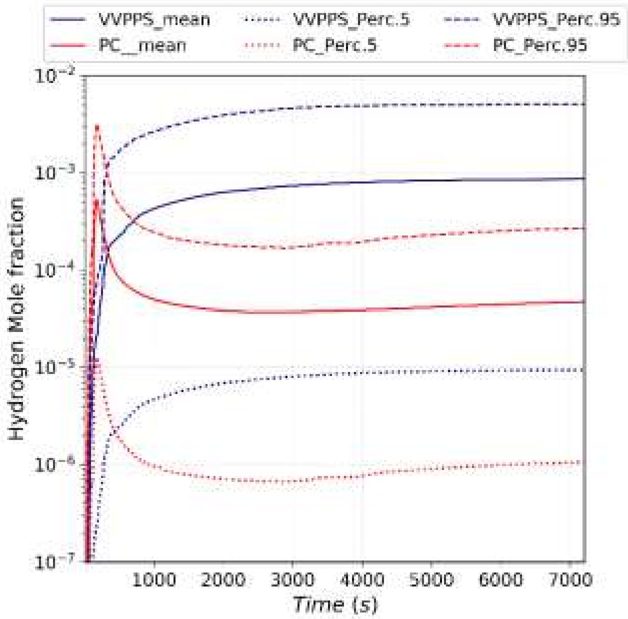

Fig. 10 Uncertainty range of hydrogen mole fraction in VVPPS and plasma volume (PC)

In figure 10 the uncertainty ranges of hydrogen mole fraction in VVPSS and plasma volume is shown. Results demonstrate that few amounts of hydrogen are produced also if the first-wall experiences very high failure. The mole fraction is well below the flammability limit of 4 % and gives a reasonable margin for model uncertainty.

8. Conclusions

A code interface has been developed by Sapienza University of Rome to couple MELCOR fusion code with RAVEN. The interface has been applied to perform a sensitivity analysis for a BDBA case scenario for the EU-DEMO reactor. The simulated accident was an ex-vessel LOCA with the additional failure of the plasma shutdown system.

Results showed that FW temperature at which plasma in-vessel breach occurs is a parameter that affects not only the mass of hydrogen produced, but also the overall VVPSS response. This study showed that if FW failure temperature is higher than 1450 K the RDs are not triggered and the VVPSS function to retain the radioactive inventory is lost. Focusing the attention on hydrogen production, it is to underline that, as expected, its uncertainty is mainly related to the temperature behavior of the FW, while the partial pressure of steam inside the VV has a low ranked influence for this accident scenario.

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