Journal of Physics: Conference Series

Severe accident sensitivity and uncertainty estimation using MELCOR and RAVEN

To cite this article: M. D'Onorio *et al* 2022 *J. Phys.: Conf. Ser.* **2177** 012021

Journal of Physics: Conference Series 2177 (2022) 012021 doi:10.1088/1742-6596/2177/1/012021

Severe accident sensitivity and uncertainty estimation using MELCOR and RAVEN

M. D'Onorio[1], A. Giampaolo[1], F. Giannetti[1], F. Mascari[2], G. Caruso[1]

[1] Sapienza University of Rome, DIAEE, Corso Vittorio Emanuele II, 244, 00186 Rome, Italy

[2] ENEA, FSN-SICNUC-SIN, Via Martiri di Monte Sole, 4 - 40129 Bologna

matteo.donorio@uniroma1.it

Abstract. Today, using a best-estimate approach is a key factor in the simulation and prediction of thermal-hydraulic and other multi-physics phenomena occurring during nuclear severe accidents. The best-estimate approach requires the quantification of both epistemic and stochastic uncertainties of safety codes to be effective. This safety assessment approach, named Best Estimate Plus Uncertainty (BEPU), is being pursued as an alternative to traditional deterministic analyses that are intrinsically conservative with not clearly defined safety margins. While in a conservative approach, the results are expressed in terms of a set of calculated conservative values of parameters, in a best-estimate methodology, the results are expressed in terms of uncertainty ranges for the calculated parameters. The International Technical Nuclear Community has made great efforts to develop methods and tools for uncertainty and sensitivity analyses of severe accident codes. In this framework, Sapienza University of Rome has developed a new Python script to use RAVEN as a tool for the application of the BEPU methods within the safety analyses performed with MELCOR. The aim of this paper is to show the capabilities of the coupling between RAVEN and MELCOR by performing a statistical analysis to estimate the range of evolution of a severe accident scenario.

1. Introduction

In evaluating safety margins, the use of the Best Estimate Plus Uncertainty (BEPU) approach by coupling selected calculated parameters with the related uncertainty range is of great interest for the International Technical Nuclear Community. Considering the reached level of development and maturity of Severe Accident (SA) codes and their application on Severe Accident Management Guideline (SAMG) assessment, the discussion and application of SA progression analyses with uncertainty estimation is currently a key topic in the BEPU framework. Several SA codes, in fact, have been developed and validated against the available experimental data in the last decades to simulate the phenomena/processes taking place in an unmitigated accident determining potential core degradation.

In this framework, MELCOR [1][2][3] is a fully integrated severe accident code able to simulate the thermal-hydraulic phenomena in steady and transient condition and the main SA phenomena characterizing the Reactor Pressure Vessel (RPV), the reactor cavity, the containment, and the confinement buildings typical of Light Water Reactors (LWR). The estimation of the source term is obtained by the MELCOR code as well. MELCOR can be used with the Symbolic Nuclear Analysis

Package (SNAP) [4] to develop the nodalization and post-process data using its animation model capabilities. MELCOR has a modular structure. Each package simulates a different part of the transient phenomenology. In particular, the Control Volume Hydrodynamics (CVH) and FLow path (FL) packages simulate the mass and energy transfer between control volumes, the Heat Structure (HS) package simulates the thermal response of the heat structure, and the CORe (COR) module evaluates the behaviour of the fuel and structures contained in the core and lower plenum and their degradation phenomena. The CVH/FL packages' role that provides the boundary condition for other packages is to be underlined.

Sapienza University of Rome has developed a new Python [5] interface to couple the SA code MELCOR and Reactor Analysis and Virtual control ENviroment (RAVEN) software tool. RAVEN, developed at the Idaho National Laboratory (INL), acts as the control logic driver and post-processing tool for different applications [6]. RAVEN has been developed to perform parametric and probabilistic analysis based on the response of complex system codes to quantify the safety margins related to safety-related events [7]. Nowadays, RAVEN is a multi-purpose probabilistic and uncertainty quantification platform that can be coupled with any system code. For this purpose, a new code interface has been developed to couple MELCOR code with the RAVEN framework.

The aim of this paper is to show the capabilities of the coupling between RAVEN and MELCOR, performing statistical analysis to estimate the range of evolution of severe accident scenarios. For this purpose, the Fukushima Daiichi Unit 1 unmitigated accident sequence has been selected as the reference scenario. A set of seven parameters have been selected among the list of modelling coefficients investigated in the SOARCA (State-of-the-Art Reactor Consequence Analyses) project [8]. Transient results have been statistically analysed through the RAVEN post-processor.

2. Plant modelling using MELCOR

Fukushima Daiichi Unit 1 is BWR/3 of 1380 MW_{th} with a steam flow rate generation in nominal operation condition of 2400 t/h. The operation pressure in RPV is about 6.99 MPa considering an average core temperature of 558 K. Fukushima Daiichi Unit 1 employed a Mark I containment with a bulb-shaped drywell and a toroidal suppression chamber, connected with radial vents and downcomers immersed in the pool. There were 8 vent lines connecting through 80 downcomers the drywell volume with the suppression chamber [9][10].

In order to develop the Fukushima unit 1 MELCOR nodalization, following the SANDIA approach reported in the Fukushima Daiichi Accident Study [11], the nodalization has been based on the Peach Bottom reactor (different power but similar reactor layout). The references used to develop the BWR Peach Bottom nodalization are [12] and [13]. Starting from this model, the Fukushima Daiichi Unit 1 input was built with the data reported by TEPCO and applying a reasonable scaling factor for each component if the data were not available. The details of the developed MELCOR input deck (geometrical and steady-state operational condition comparison with plant data) are reported in [14] and [15].

The MELCOR nodalization was designed to have a reasonable computational time and a realistic prediction of the phenomena involved during the transient, assuring a reliable and accurate transient simulation. In MELCOR nodalization, developed by using SNAP [4], the core has been modelled by coupling the CVH package nodalization with the correspondent MELCOR code model of the COR package. In order to better simulate the local heat transfer process for a wide range of fluid conditions and structure surface temperatures, each core equivalent channel has been axially split into 5 control volumes. The reactor core is modelled in the COR package with 23 axial regions and 6 radial rings. The lower plenum has been modelled with the first 13 axial regions and the core with 10 axial regions. The hydraulics control volumes of the MELCOR nodalization are shown in Figure 1.

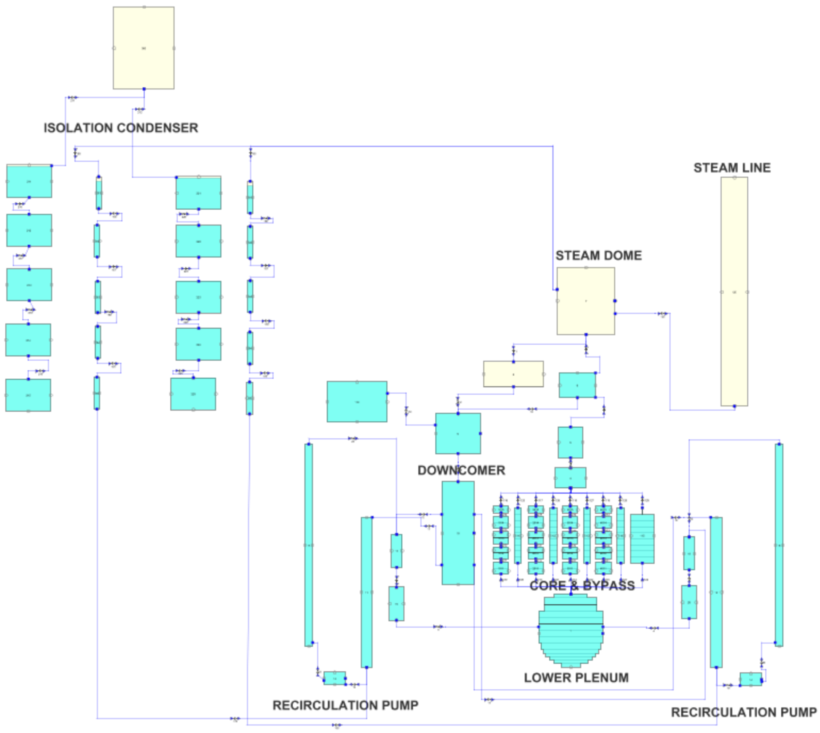

Figure 1. Thermal-hydraulic model of Fukushima Daiichi Unit 1 (made with SNAP).

3. RAVEN-MELCOR CODE COUPLING

A Python interface has been developed to allow the coupling between RAVEN and MELCOR. The interface has three manly functions, interpret the information coming from RAVEN, translate such information in the input of the driven code, and manipulate output data files to create a database. To deal with the first two functions, the "GenericCode interface" module, already implemented in RAVEN, has been coupled with a module that runs MELGEN and MELCOR executables. A Python output parser has been developed to allow RAVEN to store output data coming from MELCOR. The aim of this parser is to convert the PTF binary file generated by MELCOR into a single CSV file. The interface allows the creation of a CSV file with only plot variables required by the users to overcome the handling of large data files. So is possible to obtain a Hierarchical Data Format 5 (HDF5) database that comprises variables from all MELCOR packages (e.g., CVH, FL, COR, RN) and perform statistical analyses on the specified variables. The procedural framework used for the uncertainty quantification is shown in Figure 2.

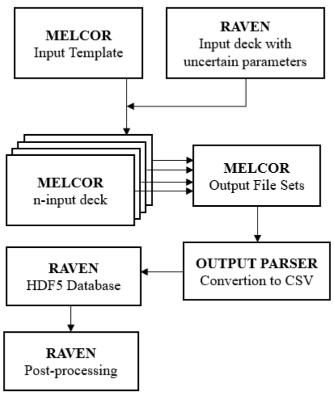

Figure 2. Uncertainty Analysis RAVEN-MELCOR flow diagram.

A MELCOR input deck is used as a template, and the chosen parameters are specified as a string with special characters. In such a way, RAVEN can identify such parameters and replace the string with values sampled from a specified distribution. Consequently, N MELCOR input-decks are generated. Once the HDF5 database has been generated, statistical analysis of the output sets can be performed.

3.1. Variables and sampling

A set of 7 parameters have been chosen to perform this study and to test the implemented code coupling between RAVEN and MELCOR. The probabilistic method to propagate input uncertainty has been selected to perform the analyses [17]. The parameters treated in this analysis are the recirculation pump seals leak flow area (plant parameter not available to the code user), and 6 MELCOR modelling parameters taken from the list of modelling parameters and sensitivity coefficient investigated in [16]. MELCOR code allows code-users to modify some important parameters model directly through code input as Sensitivity Coefficient (SC), to facilitate uncertainty analyses and sensitivity studies. The parameters selected are listed in Table 1. The Area_Seal parameter is related to the seal leak flow area. In typical BWR recirculation pumps, a combination of pressure unbalance and springs is used to keep the faces of the seal closed. If this combination is lost, a leakage from the pumps can occur. This parameter mainly affects the RPV blowdown phase and the pressurization of the DW containment. In MELCOR, the downward relocation of the core material is modeled as a constant velocity movement using a user-specified velocity of the falling debris (VFALL). Since core debris are cooled by surrounding water during the relocation phase, this velocity can affect the properties of debris relocated into the lower plenum, influencing lower head failure and melt release mode. Once debris relocates above the lower head, it increases its temperature based on debris-lower head heat transfer coefficient, input in MELCOR code as HDBLH. The default heat transfer coefficients are order-of-magnitude parameters that should be varied in sensitivity studies to determine their impact on lower head heat transfer and failure. After zirconium melting and candling by exceeding breakout temperature, UO_2/ZrO_2 eutectic reactions will form a complex mixture with a lower melting temperature than either ZrO_2 or UO_2. Temperature formation of eutectic could affect fuel failure and molten pools generation. In MELCOR the temperature at which eutectic will melt could be changed by varying the SC1132(1). The sensitivity coefficient 1141(2) governs the maximum melt Zr flow rate per unit width after breakthrough. A high flow rate corresponds to a higher mass of molten Zr relocating, thus lower local oxidation, temperature, and extended lifetime. Sensitivity coefficient 1502(2) specifies the minimum component mass below which will not be subject to the maximum temperature change criterion. In conclusion, the sensitivity coefficient 1250(1) is used to fix the temperature above which the conduction heat transfer model is enhanced where core debris is molten. This parameter is useful to capture the qualitative effects of convection in molten pools.

Once all the variables have been selected to perform an uncertainty quantification analysis, a sampling strategy needs to be employed. The sampling strategy is used to perturb the input space in relation to variable distributions. The sample size needed to obtain a significative output statistic was selected using the Wilks formula for two-sided statistical tolerance limits [18]. The required minimum number of computer code calculations becomes 93 for a 95% probability and 95% confidence level [17]. The Monte Carlo sampling strategy has been selected for this analysis, setting a limit of 250 calculations to consider possible code failures. Of the 250 MELCOR runs, 33 experienced numerical failures. The Monte Carlo method is one of the most-used sampling methodologies because it does not use a structured discretization of the input space but a cumulative probability and probability distribution function to compute the different values of the concerned variables. In table 1, the characterization of the input parameters is reported. However, this is a preliminary estimation based on engineering judgment, and it is the basis for showing the methodology used in the paper. Further studies are in progress to assess these PDFs and ranges.

Table 1. **MELCOR perturbed variables in input-deck.**

Variable	Description	Distribution	Unit	Data
Area_Seal	Recirculation pump seals leak flow area	Triangular	m^2	Mode: 0.12 E^{-3} Min: 0.06 E^{-3} Max: 0.4 E^{-3}
vfall	Velocity of falling debris	Triangular	m/s	Mode: 0.1 Min: 0.05 Max: 1.2
hdblh	Heat transfer coefficient from debris to lower head	Triangular	$\frac{\mathrm{W}}{m^2\ K}$	Mode: 1000 Min: 50 Max: 1100
SC1132(1)	Core Component Failure Parameters - Temperature to which oxidized fuel rods can stand in the absence of unoxidized Zr in the cladding.	Normal	K	Mean: 2700 Sigma: 120
SC1141(2)	Core Melt Breakthrough Candling Parameters - Maximum melt flow rate per unit width after breakthrough	Triangular	Kg/s	Mode: 0.083 Min: 0.01 Max: 1.0
SC1502(2)	Minimum Component Masses - Minimum total mass of component subject to the maximum temperature change criterion for timestep control	Normal	Kg	Mean: 10. Sigma: 1.0
SC1250(1)	Conduction Enhancement for Molten Components - Temperature above which enhancement is employed	Normal	K	Mean: 2800 Sigma: 150.0

4. Results

Results have been statistically analysed through the RAVEN BasicStatistic post-processor. A dynamic statistical analysis has been performed, setting time as a pivot parameter. To better visualize uncertainty, the 0.05, 0.5 (median) and 0.95 percentiles have been selected, analysing different figures of merits. The Pearson's and Spearman's coefficients have been used to describe the correlation between the perturbed variables and the figure of merits. In particular, Pearson's coefficient has been analysed to measure the degree of linear correlation between a perturbed variables and the figure of merits. The Spearman coefficient has been used to measure the the degree of monotonous correlation between a perturbed variables and the figure of merits. Both coefficients span between -1 and +1.

Figure 3 and Figure 4 show the spectrum of predicted DryWell (DW) and WetWell (WW) pressure, respectively. In the early stage of the accident, the gradual increase of pressure is due to steam ejection from recirculation pumps seal leak. DW pressurization is also the result of hydrogen generated from Zircalloy oxidation reaction, which production starts about 2.5 hours after the reactor SCRAM.

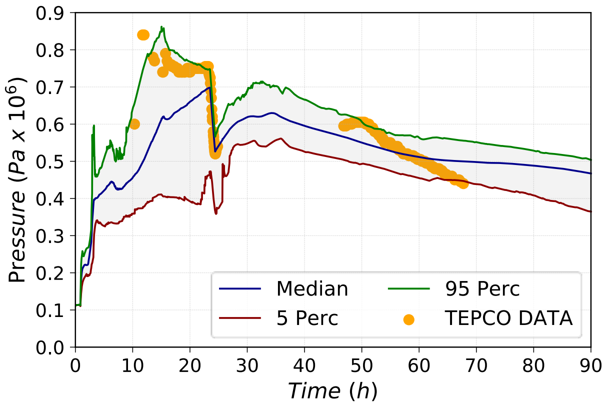

Figure 3. Variation range of Drywell pressure.

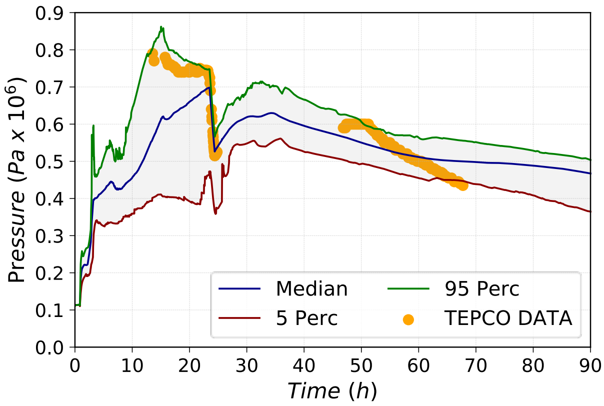

Figure 4. Variation range of Wetwell pressure.

The generated hydrogen reaches the DW and WW containment together with steam through leaks from RPV, causing a pressure increase in the containment. It is to underline that the results of the calculated data coupled with the related uncertainty band envelope mostly of the TEPCO plant data available related to both WW and DW pressure.

The pressure continued to increase until the operation of WW venting valves occurred 23 h 30 min after the earthquake. Before venting, the DW and WW pressure variation range is between 0.75 MPa and 0.44 MPa, with an expected value of around 0.7 MPa.

The venting was modelled by opening a flow path from the top of the WW to the environment. Venting activation is reported in TEPCO reference [9]. After, the venting pressure trend increases until around 33 h when the water injection with the fire engine was started.

The Pearson's and Spearman coefficient are defined between -1 and 1 and measure the linear and monotonic relationship strength between two variables. In Figure 5, the strong linear correlation between the recirculation pump seal leak area and the DW pressure is shown. The strong positive relationship highlights that as the leak area increases the pressure in containment increases, as expected. This correlation decreases with time since the long-term pressurization level of the DW depends mainly on the total inventory released and not from the leak area. Moreover, the Pearson and Spearman coefficients showed a weak correlation, lower than 0.2, between DW pressurization and other perturbed parameters.

Therefore, the lower head failure uncertainty is mainly related to the pump seal leak area, while the MELCOR modelling parameters and sensitivity coefficient, under investigation in this uncertainty application, have a low linear and ranked influence, as demonstrated by the Pearson and Spearman coefficients.

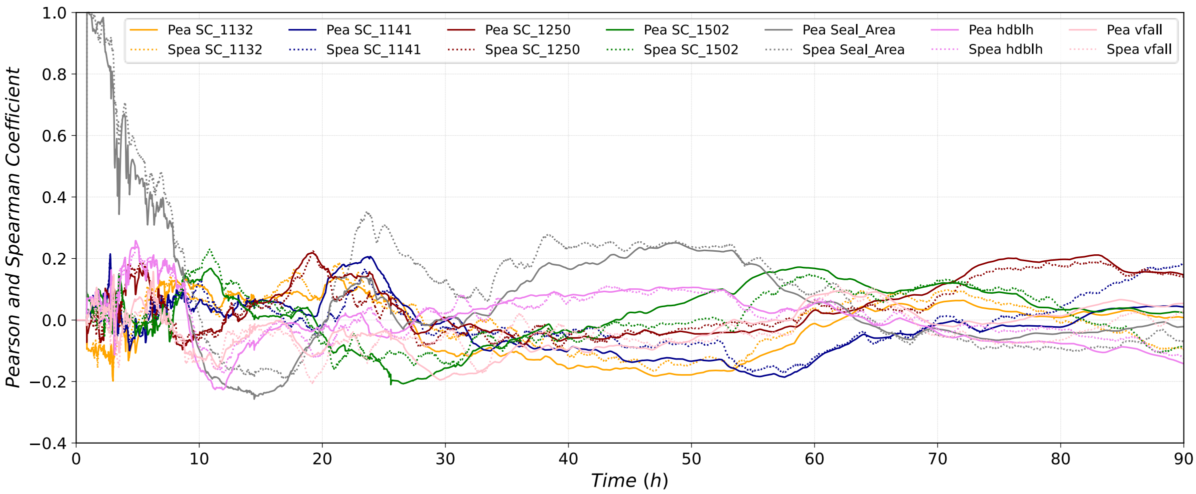

Figure 5. Pearson and Spearman coefficients for Drywell containment pressure.

The results of statistical analysis for pressure in RPV are shown in Figure 6. The first depressurization in the first hours after the SCRAM is caused by the Isolation Condenser (IC) operation. The pressure drop is predicted to start between 2 and 5.8 hours after the SCRAM and it is mainly associated with gas ejection from RPV, which can occur from the leak.

In Figure 7, the range of hydrogen mass that could be produced perturbing the input parameters is shown. At the end of calculations, a median value of 665 kg was produced, with a total mass ranging from 550 kg and 755 kg of quantiles. The onset of hydrogen production is not particularly affected by the perturbed parameters. From the output data analysis, SC1141(2) has a moderate linear correlation with hydrogen generation. Instead, the Area_Seal parameter has a weak negative correlation because its increment reduces the amount of steam available for the oxidation reaction.

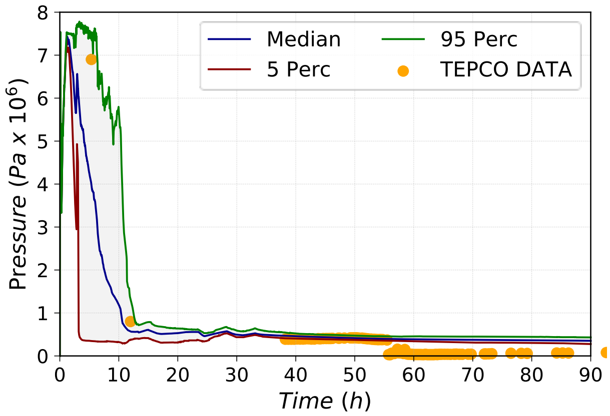

Figure 6. Variation range of RPV pressure.

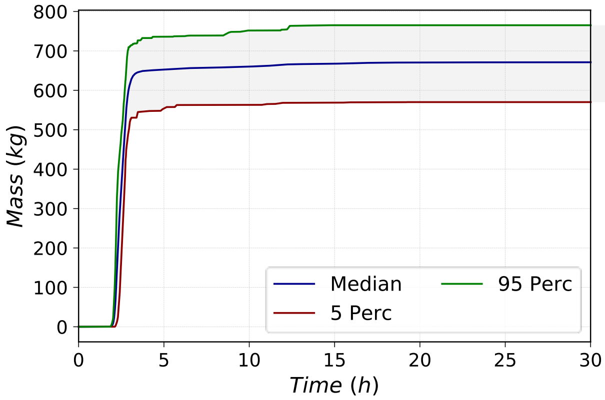

Figure 7. Variation range of mass of H_2 generated.

The Pearson and Spearman correlation coefficients have also been used for RPV pressure to evaluate the linearity and the correlation between the variables and the selected figures of merit during the transient, respectively. The time-dependent analysis is important to show how the correlation evolves during the phase of the transient. Figure 8 shows that the Area_seal is linear and strongly correlated with the RPV pressure trend in the first part of the transient. However, in this case, the trend is negative; in fact, an increase in leak area will cause a decrease in RPV pressure, as expected. Therefore, the RPV pressure is mainly related to the pump seal leak area uncertainty, while the MELCOR modelling parameters and sensitivity coefficient, under investigation in this uncertainty application, have a low linear and weak correlation, as demonstrated by Pearson and Spearman coefficient values, respectively. It is to be underlined that the results of the calculated data coupled with the related uncertainty band contain mostly available TEPCO plant data related to the RPV pressure, Figure 6.

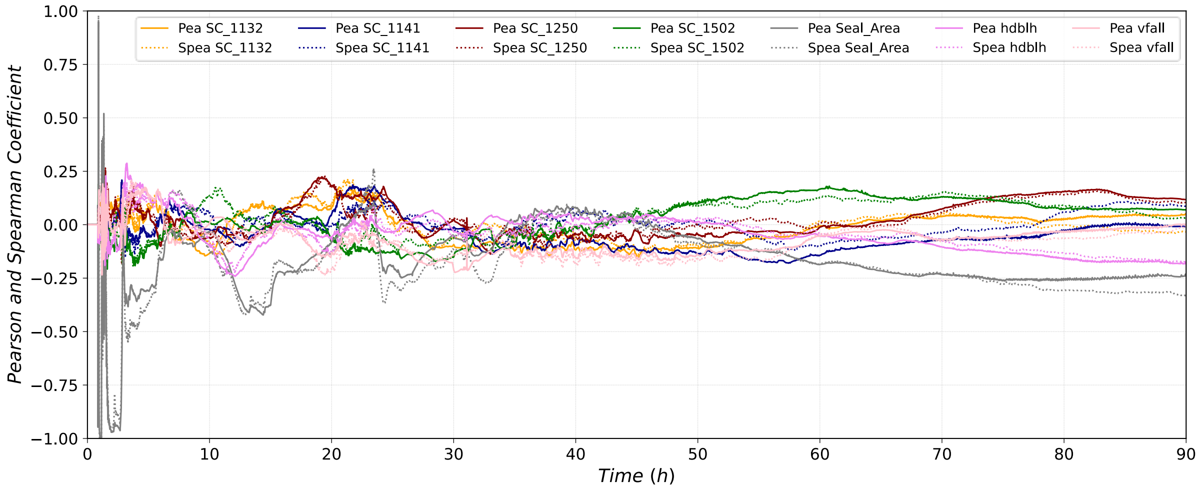

Figure 8. Pearson and Spearman coefficients for RPV pressure

5. Conclusions

The implemented code coupling between RAVEN and MELCOR has been successfully tested in the uncertainty analysis applied to the MELCOR simulation of the Fukushima Daiichi unit 1 severe accident progression. The present analysis highlights some uncertainty that produces differences in the timing of the major events, such as lower head failure and pressure behaviours. Considering the uncertainty band chosen (0.05, 0.5, and 0.95 percentiles), publicly available TEPCO plant data selected for this analysis are generally contained on it. Sensitivity analyses are in progress to study the effect of the nodalization details on the calculated data. Regarding the lower head failure time, it is to underline that there is a linear and strong correlation with the pump seal leak area. At the same time, the MELCOR modelling parameters and sensitivity coefficients, under investigation in this application, have a low linear and weak correlation, as demonstrated by Pearson and Spearman coefficients. Same considerations can be made for the pressure behaviours here investigated. In the future, a more detailed analysis will be performed, through the models implemented in RAVEN, by increasing the number of variables affected by uncertainty and including the uncertainty of the TEPCO plant data available in the literature.

References

[1] Humphries LL, Beeny BA, Gelbard F, Louie DL and Phillips J 2015 *MELCOR Computer Code Manuals, Vol. 1: Primer and Users' Guide, Version 2.1.6840*, SANDIA REPORT SAND-2015-6691R, Sandia National Laboratories, Albuquerque, New Mexico. https://www.nrc.gov/docs/ML1704/ML17040A429.pdf

[2] Humphries LL, Beeny BA, Gelbard F, Louie DL and Phillips J 2017 *MELCOR Computer Code Manuals, Vol. 2: Reference Manual,* Version 2.1.6840, SANDIA REPORT SAND2017-0876O, Sandia National Laboratories, Albuquerque, New Mexico. https://www.nrc.gov/docs/ML1704/ML17040A420.pdf

[3] Pescarini M, Mascari F, Mostacci D, De Rosa F, Lombardo C and Giannetti F 2017 *J. Phys.: Conf. Ser*. **923** 012009.

[4] Symbolic Nuclear Analysis Package (SNAP) 2021 Users Manual. Applied Programming Technology, Inc. Bloomsburg, PA

[5] G. van Rossum, Python tutorial, Technical Report CS-R9526, Centrum voor Wiskunde en Informatica (CWI), Amsterdam, May 1995

[6] Rabiti C, Alfonsi A, Cogliati J, Mandelli D, Kinoshita R, Sen S, Wang C, Talbot WP, Maljovec DP and Chen J 2017 *RAVEN User Manual,* INL/EXT-15-34123. https://inldigitallibrary.inl.gov/sites/sti/sti/Sort_4450.pdf

[7] Rabiti C, Alfonsi A, Cogliati J, Mandelli D and Martieneau R 2012 *RAVEN as Control Logic and Probabilistic Risk Assessment Driver for RELAP-7*, Proceeding of American Nuclear Society (ANS), 107, pp 333–335, https://www.osti.gov/servlets/purl/1060968

[8] Bixler NE, Ross K, Sallaberry C, Jones JA and Ghosh ST 2016. *State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Short-Term Station Blackout of the Surry Power Station*. Available at https://www.osti.gov/servlets/purl/1380180

[9] TEPCO 2011 Fukushima Nuclear Accident Analysis Report. Tokyo: Tokyo Electric Power Company, Inc. December 2. Available at http://www.tepco.co.jp/en/press/corp-com/release/betu11_e/images/111202e14.pdf

[10] TEPCO, Plant Specifications of Unit 1, https://fdada.info/en/home2/accident2/specification2/

[11] Gauntt RO, Kalinich D, Cardoni J, Phillips J, Goldmann A, Pickering S, Francis M, Robb K, Ott L, Wang D, Smith C, Germain S, Schwieder D and Phelan Ch 2012. *FUKUSHIMA DAIICHI: ANS Committee Report, Fukushima Daiichi Accident Study (Status as of April 2012).* SANDIA REPORT SAND2012-6173 https://doi.org/10.2172/1055601

[12] Bixler N, Gauntt R, Jones J and Leonard M 2013 *State-of-the-Art Reactor Consequence Analyses Project Volume 1: Peach Bottom* Integrated Analysis. NUREG/CR-7110, Vol. 1, https://www.nrc.gov/docs/ML1315/ML13150A053.pdf

[13] Carbajo JJ 1993, *Severe Accident Source Term Characteristics for Selected Peach Bottom Sequences Predicted by the MELCOR Code*, NUREG/CR-5942, https://doi.org/10.2172/10182846

[14] Mascari F, Pescarini M, Giannetti F, Luciani I and Caruso G *Integral Calculations of Severe Accident Scenarios in PWR and BWRs*, ADPFISS-LP1-075, 2016

[15] Mascari F, D'Onorio M, Giannetti F, Caruso G and Naviglio A., *Analyses of unmitigated transients. Small break LOCA in PWR, Loss of Feed Water in PWR and SBO in BWR,* Report ADPFISS-LP1-103
[16] Ross K, Phillips J, Gauntt RO and Wagner KC 2014, *MELCOR Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project*, NUREG/CR-7008, U.S. NRC. https://www.nrc.gov/docs/ML1423/ML14234A136.pdf
[17] Glaser H 2008 *GRS Method for Uncertainty and Sensitivity Evaluation of Code Results and Applications.* Science and Technology of Nuclear Installations, 798901, https://doi.org/10.1155/2008/798901
[18] Wilks SS 1941 *Determination of sample sizes for setting tolerance limits* Ann. Math. Stat., 12, pp. 91-96. https://doi.org/10.1214/aoms/1177731788
[19] ICANPS. (2012). Final Report Investigation Commitee on the Accident of Fukushima Nuclear Power Stations of Tokyo Electric Power Company. Available at https://www.cas.go.jp/jp/seisaku/icanps/eng/final-report.html