Italian National Agency for New Technologies, Energy and Sustainable Economic Development

ENEA ACTIVITIES IN THE FIELD OF SEVERE ACCIDENTS

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14th European MELCOR User Group (EMUG), Ljubljana, (Slo), 12– 14 April, 2023

INTRODUCTION

- ENEA activities in the field of severe accidents by using MELCOR code is based on the analyses of current and advanced reactor designs in steady and transient conditions.

- The following activities are developped:
  - Simulation and evaluation of severe accident evolutions and source term for “safety assessment”, mainly focusing on the characteristics of NPPs located at the Italian border;
  - Code to code benchmark with other severe accident codes (e.g. ASTEC, MAAP, etc);
  - Analyses of MELCOR code capability against experimental data for current and advanced designs;
  - Coupling of MELCOR code with uncertainty tools (e.g. DAKOTA, RAVEN, etc).
  - Use of the MELCOR code for fusion technology applications.

- The activities are developped in International, European and Italian research projects:
  - European Union Research and Innovation Programme (e.g. SASPAM-SA, MUSA,..);
  - NUGENIA project (e.g. ASCOM,..);
  - IAEA CRP;
  - EUROFUSION activity;
  - Italian domestic project funded by the Ministry of Economic development;
  - Etc

- The activities are developped in collaboration with Italian Universities.

ENEA

ENEA ACTIVITIES IN THE FIELD OF SEVERE ACCIDENTS BY USING MELCOR CODE

- MELCOR/DAKOTA COUPLING IN A SNAP ENVIRONMENT ARCHTECTURE
  - Considering that currently the International Nuclear Technical Community is exploring the possibility of using severe accident code in a BEPU framework (e.g. MUSA, IAEA CRP(I31033), etc) , the target of this activity is to show the main details and capability of the MELCOR/DAKOTA coupling in a SNAP environment/architecture, and the different steps necessary to set-up it.
    - ***A NUREG-IA is going to be published to investigate the MELCOR/DAKOTA coupling in a SNAP environment/architecture***
    - *A MELCOR/DAKOTA COUPLING group has been established in the CSARP framework to be a platform of discussion and two meetings have been already organized.*
  - Activities are on going

- EU project MUSA (MANAGEMENT AND UNCERTAINTIES OF SEVERE ACCIDENTS):
  - ENEA is involved in the development of a PHEBUS FPT1 uncertainty analyses by using MELCOR/DAKOTA coupling in a SNAP environment architecture and in a Python environment/architecture;
  - ENEA is involved in the development of PWR-900 uncertainty analyses by using MELCOR/DAKOTA coupling in a Python environment/architecture;
  - ENEA is the leader of:
    - WP4: Application of UQ Methods against Integral Experiments (AUQMIE);
    - Sub-WP3.3: Feedback integration from application of uncertainty tools;
    - Sub-WP6.3: Assessment of potential radiological consequences reduction from innovative SAM measures and systems;
    - Has coordinate the DAKOTA review and is the leader of MELCOR for the elaboration of guidelines for the use of UaSA codes/methods (WP3.2).
  - The project is on going

ENEA ACTIVITIES USING MELCOR CODE

- Development of a Design 2 MELCOR input-deck in the frame of SASPAM-SA:
    - The objective of the current input-deck is to analyze the code capability and the consequent applicability to model an integral-type reactor and to simulate the complex thermal–hydraulic phenomena occurring in a passive mitigation strategy.

- ENEA INVOLVMENT IN EUROFUSION: Accident analyses within EUROFusion project:
    - DEMO Tokamak plant: Divertor system analyses
    - DONES neutron Irradiation facility
    - Activities are on going

- NUGENIA ASCOM (ASTEC COMMUNITY) Project:

ENEA is involved in the:

- WP4: ASTEC analyses at plant scale and associate crosswalk studies: here MELCOR code is used to benchmark ASTEC code:
    - A comparison for an unmitigated SBO has ben conducted in the 2019;
    - A comparison for an unmitigated 2-inch Cold Leg LOCA accident has ben conducted in the 2020

- ENEA ACTIVITIES IN THE FIELD OF SEVERE ACCIDENTS (FUSION REACTOR): activity developed in the EMUG framework:
    - Analyses of the physical models necessary to be implemented in MELCOR_2.2 for fusion reactor safety analyses and the current models already implemented in MELCOR fusion
    - A paper named “Current status of MELCOR 2.2. for fusion safety analyses” will be presented at the SOFT 2022 conference

ENEA ACTIVITIES IN THE FIELD OF SEVERE ACCIDENTS

- Development of a QUENCH-06 MELCOR input-deck:
  - Activity developed in the framework of IAEA CRP I31033.
  - Indipendent user validation of Core Heat up, Zircaloy-Steam Oxidation and Degradation models embedded into best estimate MELCOR code employing the experimental dataset provided by QUENCH-06 test;
  - Sensitivity analysis (SA) adopting several Zircaloy-Steam oxidation reaction rates;
  - Uncertainty analysis (UA).

- Sensitivity analysis on MELCOR accumulators modelling:
  - Analysis of the available accumulators models in MELCOR simulating an unmitigated SBLOCA in a generic PWR 900 like.
  - A paper will be presented at ERMSAR2022.

- Participation within Italian domestic project funded by the Ministry of Economic development.
  - In the framework of the ENEA-MSE agreement, ENEA activities related to the MELCOR code are mainly oriented to the evaluation of severe accident source term for "safety assessment activity", mainly focusing on the characteristics of NPPs located near the Italian border.
  - The project is finished.

ENEA PROJECT FINISHED IN THE FIELD OF SEVERE ACCIDENTS

- EU project IVMR (In-Vessel Melt Retention Severe Accident Management Strategy for Existing and Future NPPs):
  - ENEA has been involved in the development of a "PWR 900 like" input-deck with MELCOR code for benchmarking ASTEC code in relation to the In-Vessel Melt Retention issues during an unmitigated SBO.
  - A first calculation phase and a second phase for revised calculation have been performed.
  - The project is finished.

- EU-FASTNET project (FAST Nuclear Emergency Tools):
  - ENEA has been involved in the development of a source term database with MELCOR for selected transients.
  - The project is finished

- EU project JASMIN (Joint Advanced Severe Accidents Modelling and Integration for Sodium-Cooled Fast Neutron Reactors)
  - ENEA used the MELCOR code in order to benchmark the CPA module of ASTEC-NA.
  - The project is finished.
  - *A NUREG-IA based on the MELCOR calculation has been proposed; the proposal has been accepted and the NUREG-IA analyses should be based on the update code prediction obtained with the MELCOR release where the pool fire modeling have been implemented.*

- EU-CESAM project (Code for European Severe Accident Management):
  - ENEA has been involved in the development of a "PWR 900 like" input-deck with MELCOR code for benchmarking the ASTEC code.
  - The project is finished.
  - A NUREG-IA has been published: F. Mascari, J. C. De La Rosa Blul, M. Sangiorgi, G. Bandini, Analyses of an Unmitigated Station Blackout Transient in a Generic PWR–900 with ASTEC, MAAP and MELCOR Codes (NUREG/IA-0515), U.S. Nuclear Regulatory Commission Washington, DC 20555-0001

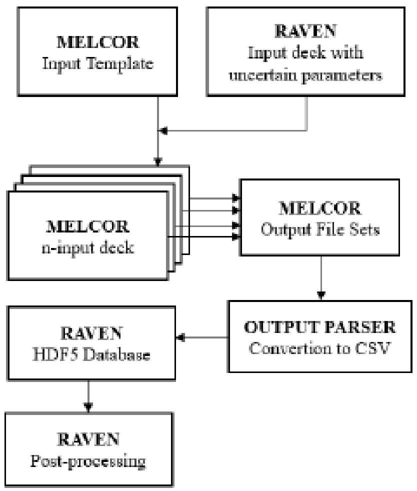
THANK YOU FOR YOUR KIND ATTENTION

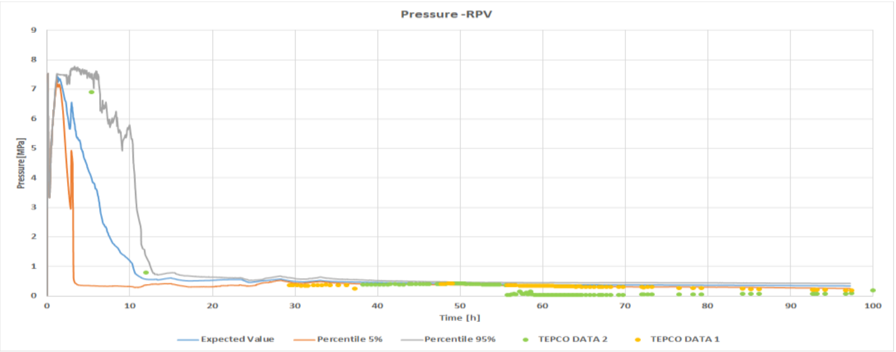
ENEA ACTIVITIES IN THE FIELD OF SEVERE ACCIDENTS – UNCERTAINTY ANALYSES

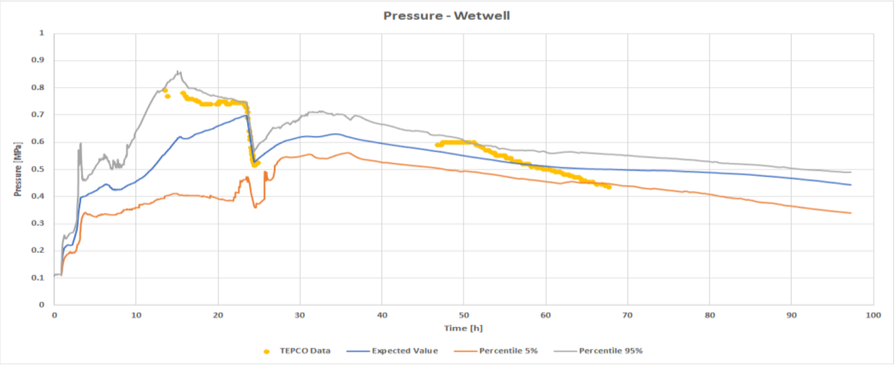
- In the evaluation of safety margins, the use of BEPU approach by coupling selected calculated parameters with the related uncertainty range is of great interest for the International Scientific Community.
- Considering the reached level of development and maturity of severe accident codes and their application on SAMG assessment, the discussion and application of severe accident progression analyses with uncertainty estimation is currently a key topic in BEPU applications.
- In the view of the research activity that are currently under development in domestic and in international framework *(e.g. MUSA project funded in the H2020 European Framework Programme, etc, IAEA CRP, etc),* ENEA have started different activities related to the uncertainty estimation:
  - Developing uncertainty analyses using the DAKOTA software tool coupled with MELCOR code in SNAP environment/architecture;
  - In collaboration with Sapienza University of Roma is developing uncertainty analyses using the RAVEN software tool coupled with MELCOR code;
  - In collaboration with University of Palermo is developing uncertainty analysis using the DAKOTA software tool coupled with MELCOR code in a Python environment/architecture;
  - In collaboration with University of Palermo is developing an in-house Python based tool for developing uncertainty analysis with MELCOR;
  - In collaboration with University of Palermo is developing uncertainty analyses using the DAKOTA software tool coupled with TRACE code in SNAP environment/architecture (TH activity);
  - In collaboration with IRSN and University of Bologna is developing uncertainty analyses using RAVEN software tool coupled with ASTEC code.
  - In collaboration with UNIPI is developing the Python Script Stream in SNAP to manage the MELCOR runs failure.

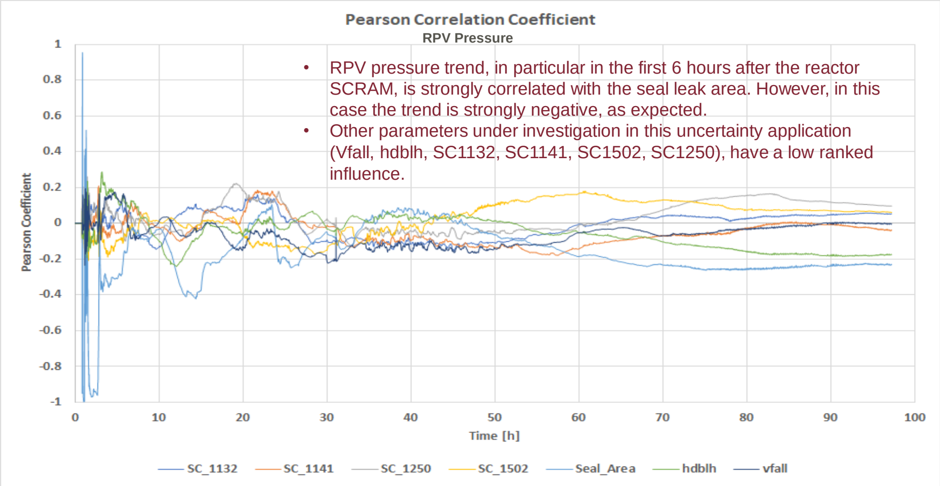
UNCERTAINTY ANALYSES

MELCOR/RAVEN:
Analysis of the BWR FUKUSHIMA DAIICHI UNIT 1 SEVERE ACCIDENT







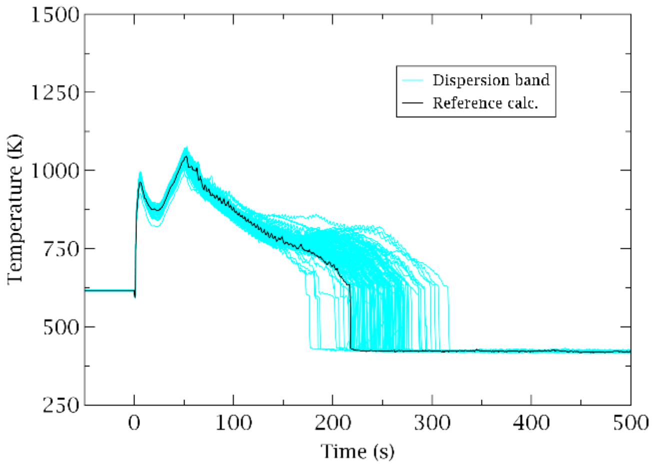


- RPV pressure trend, in particular in the first 6 hours after the reactor SCRAM, is strongly correlated with the seal leak area. However, in this case the trend is strongly negative, as expected.
- Other parameters under investigation in this uncertainty application (Vfall, hdblh, SC1132, SC1141, SC1502, SC1250), have a low ranked influence.

The activity has been done in collaboration with Sapienza University of Rome.

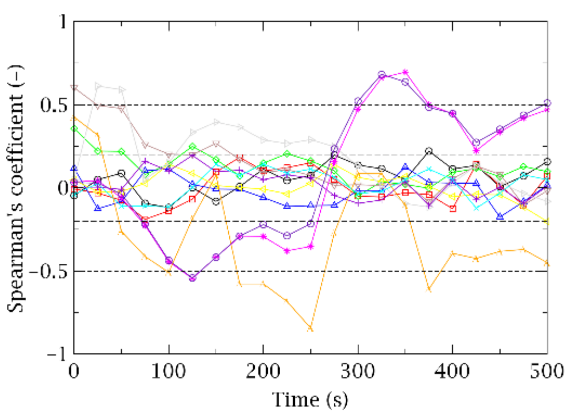
UNCERTAINTY ANALYSES

TRACE/DAKOTA:
COLD LEG LBLOCA UNCERTAINTY ANALYSIS USING TRACE/DAKOTA COUPLING



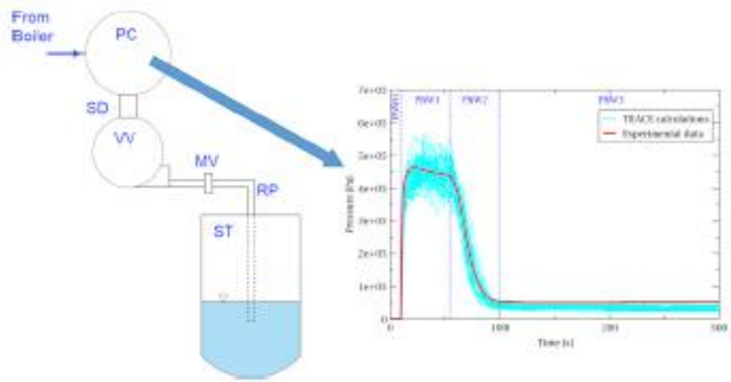
Cladding temperature of the hot rod predicted by TRACE code

Spearman rank correlation coefficient



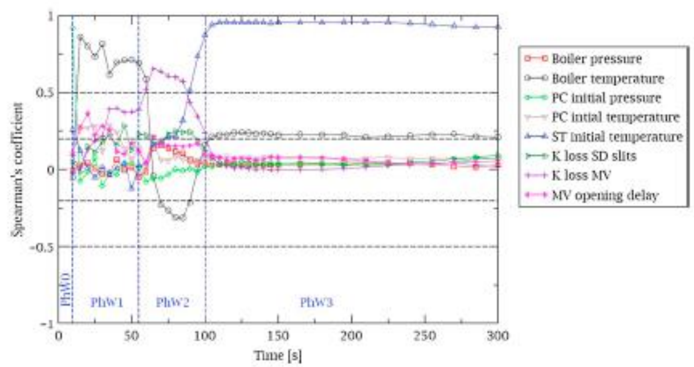
The activity has been done in collaboration with University of Palermo.

TRACE/DAKOTA:
INGRESS OF COOLANT EVENT SIMULATION WITH TRACE CODE WITH ACCURACY EVALUATION AND COUPLED DAKOTA UNCERTAINTY ANALYSIS



PC pressure predicted by TRACE code

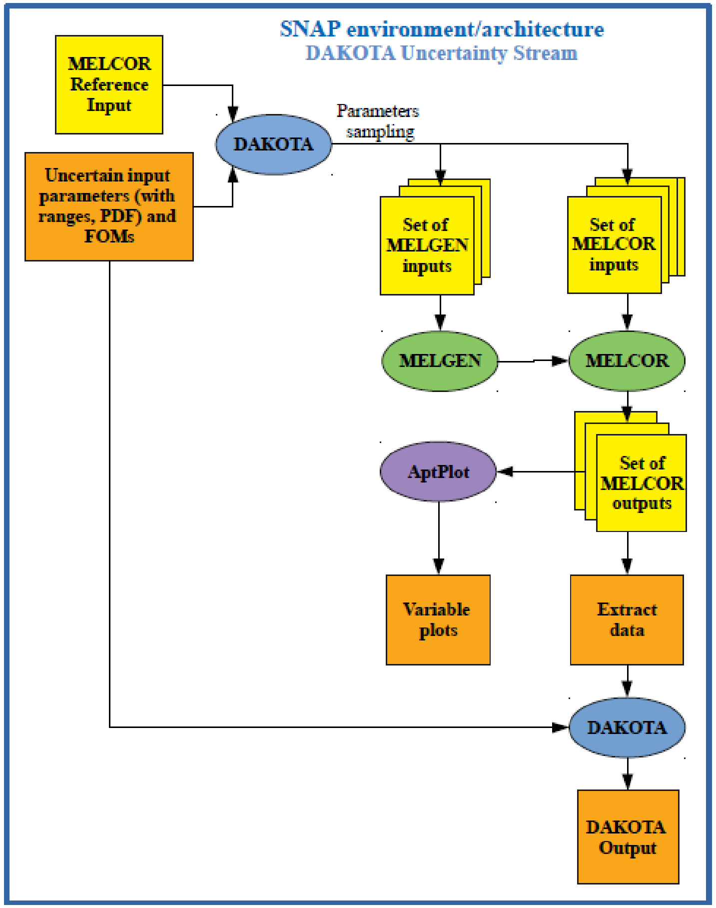
Spearman rank correlation coefficient



The activity has been done in collaboration with Politecnico di Torino and University of Bologna.

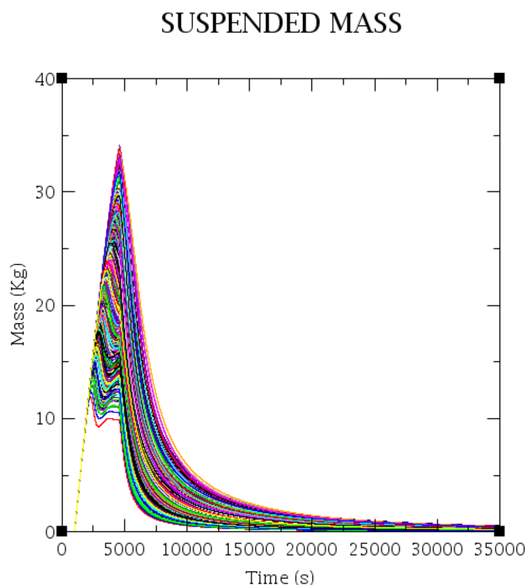
MELCOR – DAKOTA COUPLING FOR UNCERTAINTY ANALYSES, IN A SNAP ENVIRONMENT/ARCHITECTURE

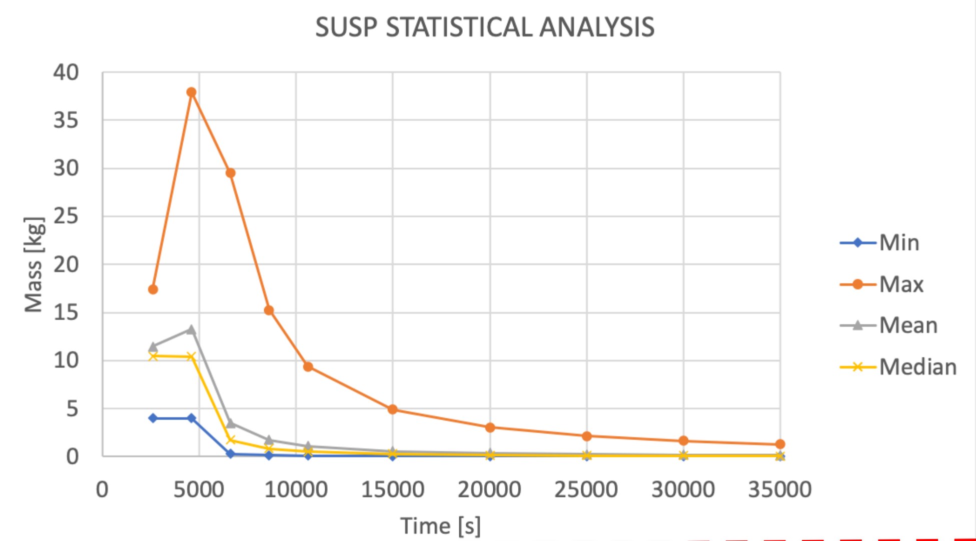
- Considering that currently the International Community (e.g. MUSA, IAEA CRP (I31033), etc) is exploring the possibility of using SA code in a BEPU framework, the target of this activity is to test and show the main details and capability of the MELCOR/DAKOTA coupling in a SNAP environment/architecture and the different steps necessary to set-up it.
- A NUREG–IA, developed together with SANDIA, UNIPA and UNIPI, has been submitted for publication and has the main target of:
  - Showing the main details of the MELCOR/DAKOTA coupling in a SNAP environment/architecture;
  - Showing the different steps necessary to set it up;
  - Describing two sample applications to show the feasibility and to analyze the capabilities of this coupling (*these first excercises aiming to show only the complete application of the coupling procedure of MELCOR and DAKOTA in a SNAP enviroment/architecture; they do not want to represent a complete and representative analyses of the MELCOR code uncertainty).*
  - Supporting MECOR users.

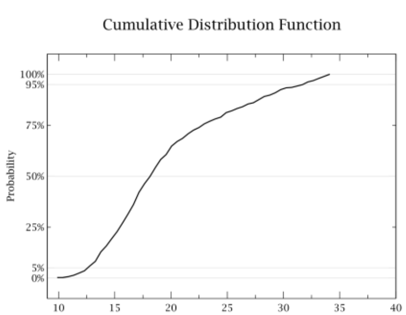


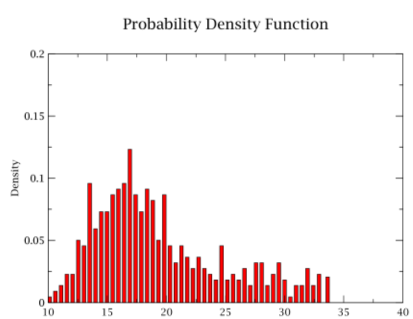
MELCOR – DAKOTA COUPLING FOR UNCERTAINTY ANALYSES, IN A SNAP ENVIRONMENT/ARCHITECTURE

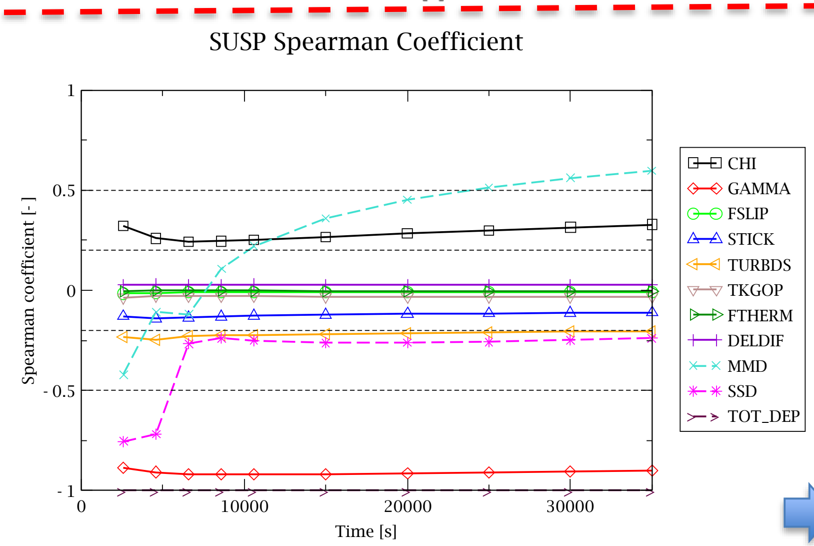
MELCOR/DAKOTA:
MELCOR/DAKOTA COUPLING AGAINST CSTF-AB1 test











A NUREG-IA has been prepared to investigate the MELCOR/DAKOTA coupling in a SNAP environment/architecture and to be an endorsed user-guide

NUREG/IA-

International Agreement Report

MELCOR – DAKOTA Coupling for Uncertainty Analyses, in a SNAP Environment/Architecture

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UNDER PUBBLICATION

A NUREG-IA has been prepared to investigate the MELCOR/DAKOTA coupling in a SNAP environment/architecture and to be an endorsed user-guide

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The activity is done in collaboration with SANDIA, UNIPA and UNIPI. A section related to the use of the use of Python directed job-stream for managing the failed calculation can be added based on the UNIPI contribution

ENEA

EU-MUSA PROJECT

- Management and Uncertainties Of Severe Accidents (MUSA) project was founded in HORIZON 2020 EURATOM NFRP-2018 call on “Safety assessments to improve accident management strategies for generation II and III reactor, and it is coordinated by CIEMAT (Spain).

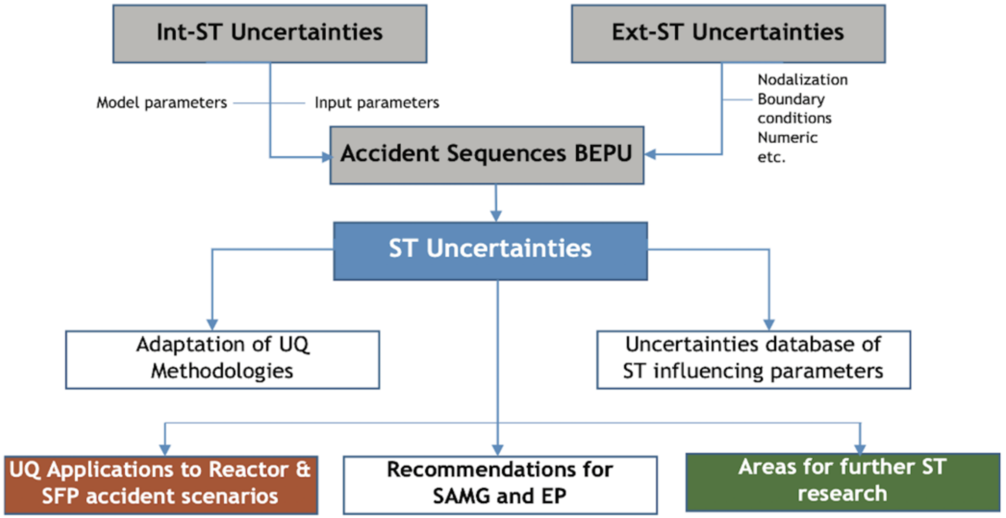
- The project started on the 1st June 2019 and the planned duration is 48 months; the overall cost is 5.768,452.50.

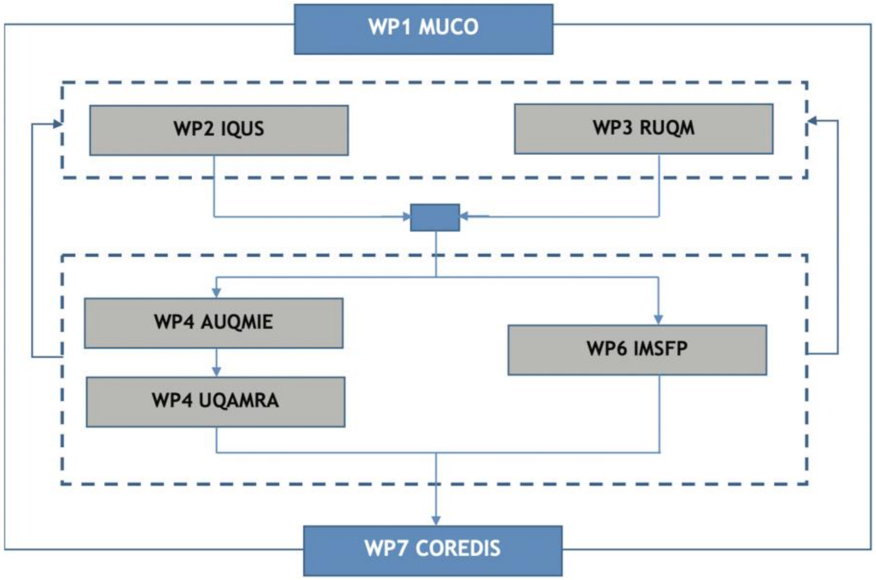
- 28 Organizations from 16 Countries are involved, and it has the NUGENIA label that recognizes the excellence of the project proposal (obtained on 7 July 2018).

- MUSA project aims to establish a harmonized approach for the analysis of uncertainties and sensitivities associated with Severe Accident (SA) analysis among EU and non-EU entities.

- The main objective of the project is to assess the capability of SA codes when modelling Nuclear Power Plant (NPP)/ Spent Fuel Pool (SFP) accident scenarios of GEN II, GEN III designs through the:
  - Identification of Uncertainty Quantification (UQ) methodologies to be employed, with emphasis on the effect of both existing and innovative SA Management (SAM) measures on the accident progression, particularly those measures related to the Source Term (ST) mitigation;
  - Determination of the state-of-the-art prediction capability of SA codes regarding the ST that potentially may be released to the external environment, and to the quantification of the associated code’s uncertainties applied to SA sequences in NPPs and SFPs.

EU-MUSA PROJECT





EU-MUSA PROJECT

- WP4, named AUQMIE (Application of UQ Methods against Integral Experiments), is aimed at applying and testing UQ methodologies, investigated in the WP3, lead by KIT, against the internationally recognized PHEBUS FPT1 test.

- The WP4, lead by ENEA, is divided in three main subWPs: the specification phase (WP4.1) lead by IRSN, the calculation phase (WP4.2) lead by GRS, and the analyses of the results (WP4.3), lead by UNIPI.

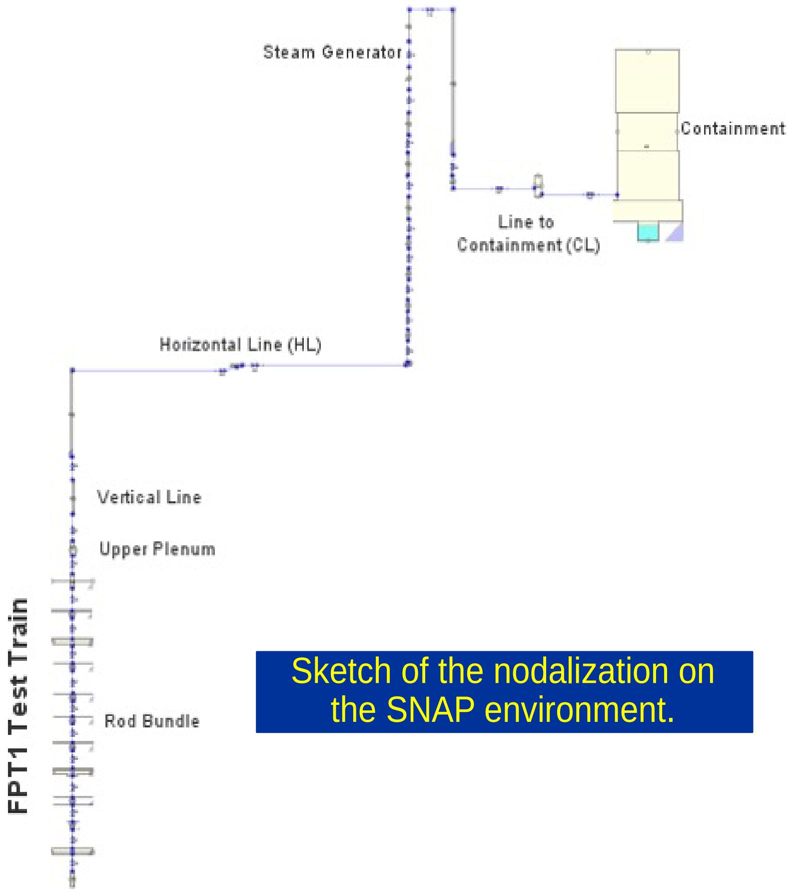
- Partners involved are CIEMAT, CEA, CNPRI, CNSC, CNPE, ENEA, LLC ENERGORISK, EPRI, GRS, INRNE, IRSN, KIT, LEI, NPIC, PSI, SNERDI, TRACTEBEL, SSTC NRS, TUS, UNIPI, UNIRM1, USNRC, VTT.

- This UQ application:
  - Will train the project Partners gaining experience in the Uncertainty and Sensitivity Analyses (UaSA);
  - Will also provide a platform of discussion for proposing solutions if some issues arise during the UaSA applications and
  - Will be used as a technical background for the full plant, WP5 lead by JRC, and SFP, WP6 lead by IRSN, UaSA application.

- The selection of the Figure of Merits (FOMs) of the uncertainty analyses (e.g., release of iodine from the test fuel bundle, amount of suspended iodine in the containment's atmosphere, etc) and of the main uncertainty sources have been done in WP2, lead by GRS.

ENEA APPLICATIONS IN EU-MUSA PROJECT

- Since Italy is a member of NRC's Cooperative Severe Accident Research Program (CSARP), ENEA has requested a PHEBUS FPT1 input-deck to USNRC. USNRC disclosed it and granted permission to ENEA to use it as a part of international collaboration on the MUSA project.
- Based on the USNRC input-deck, the SNAP model has been developed.
- The nodalization of the Phebus FPT1 used for the following study is composed of 31 control volumes, 29 Flow Paths and 68 Heat Structures.
- The fuel bundle is axially subdivided into eleven control volumes, modelled with the MELCOR CVH package. The vertical line above the bundle is subdivided into three regions corresponding to the Upper Plenum (UP), the lower vertical line and the upper vertical line.
- The horizontal line (HL) connecting vertical line above the bundle and inlet of the Steam Generator (SG) is modelled with two control volume. The SG is characterized by 9 control volumes in the ascending side and two control volumes in the descending side. There is one control volume connecting the outlet of the SG to the containment, which is modeled with one control volume.
- The bundle test section is modelled, in the COR package, by 31 axial regions and 2 radial regions.



Sketch of the nodalization on the SNAP environment.

ENEA APPLICATIONS IN EU-MUSA PROJECT

- The following FOMs has been investigated in the reference case:

  - Release of iodine from the test fuel bundle [% of initial inventory];
  - Release of Caesium from the test fuel bundle [% of initial inventory];
  - Caesium retention in the circuit [%of Cs released from the bundle];
  - Aerosol mass in suspension in the containment’s atmosphere [g];
  - Amount of suspended iodine in the containment’s atmopshere [g];
  - Total deposited iodine in the containment [g].

- The calculation time is: 29500 s.

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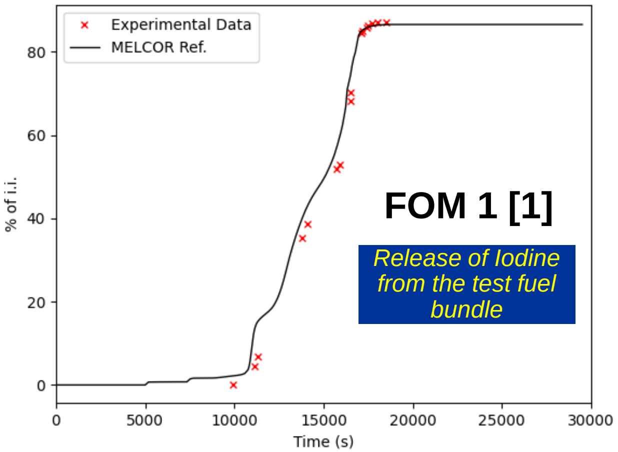
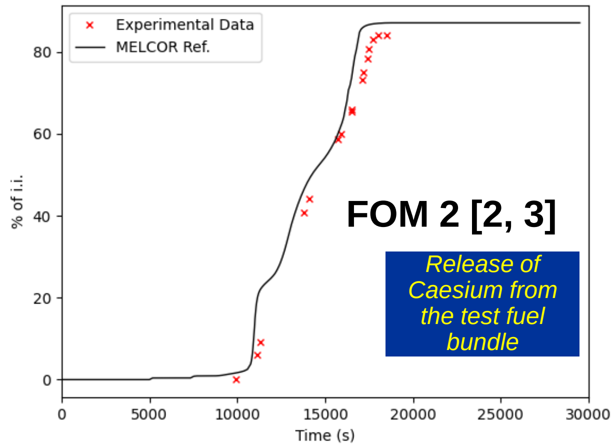
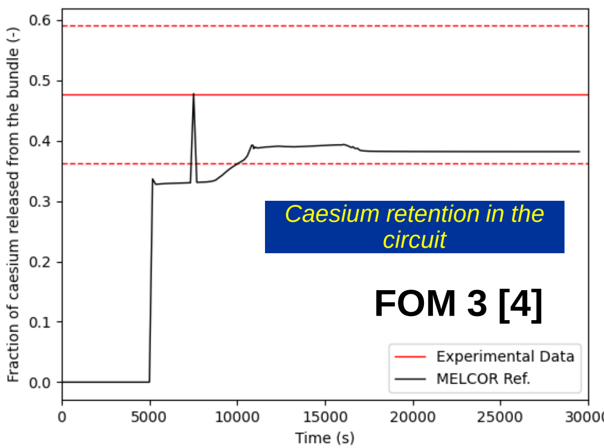
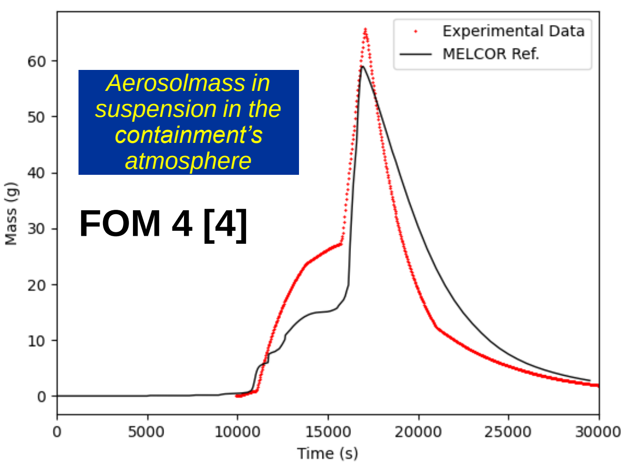
ENEA APPLICATIONS IN EU-MUSA PROJECT

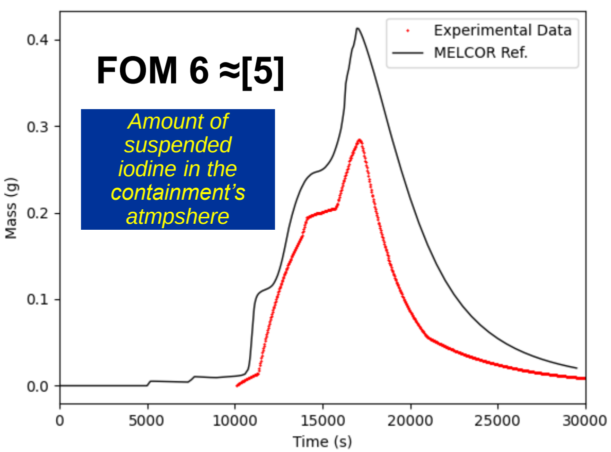
Analyses of the code accuracy to predict the MUSA-WP4 FOMs in the reference case

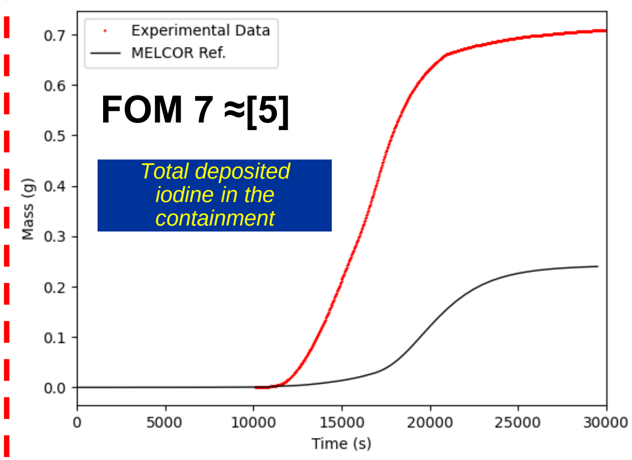






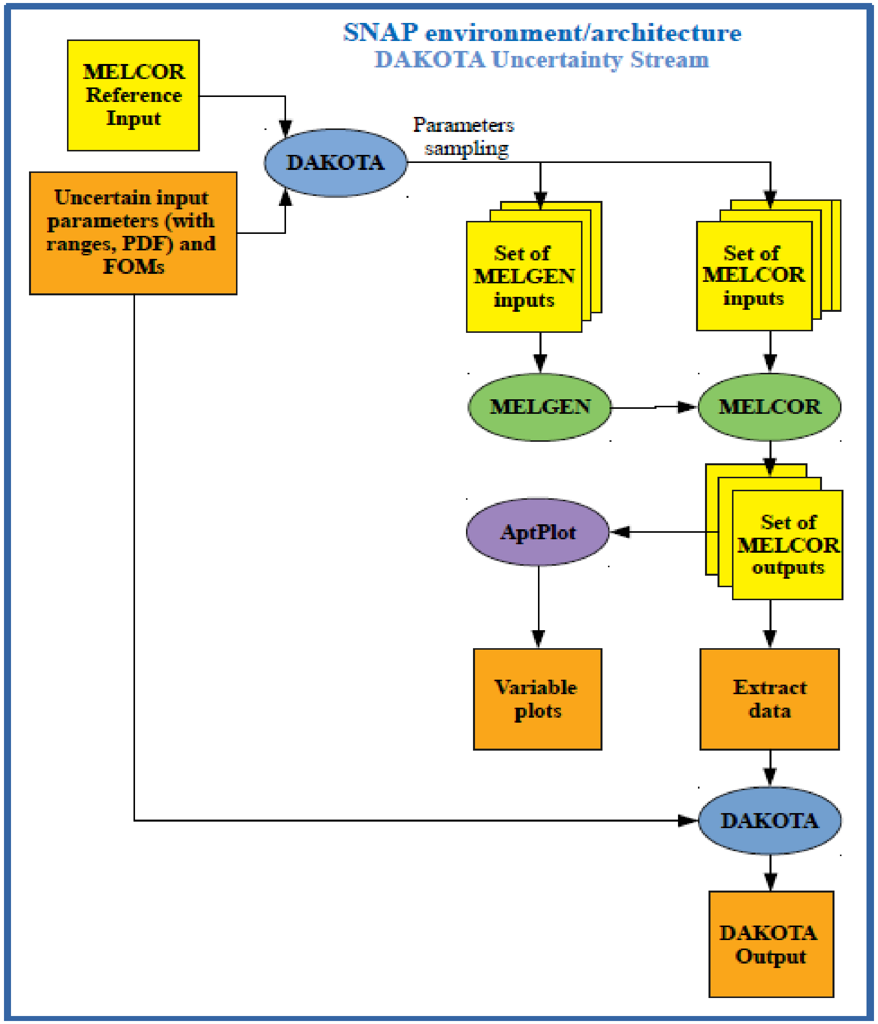






ENEA APPLICATIONS IN EU-MUSA PROJECT

- To develop the uncertainty analysis of this case, the coupling MELCOR/DAKOTA in SNAP environment/architecture has been considered.
- DAKOTA toolkit, provide as a plug-in for SNAP, allows the following:
  - Enter the uncertain input parameters (with range and PDF);
  - Select the sampling method (Monte Carlo or Latin Hypercube);
  - Enter the desired FOMs for the analysis;
  - Set the final report.
- DAKOTA is used both at the beginning of the analysis to sample the uncertain input parameter values and to generate the set of code inputs.
- Then, after the solution of the set of code inputs and the extraction of the desired data, DAKOTA performs the uncertainty analysis and apply regression techniques to evaluate the correlation between input and output parameters selected as a FOMs.
- At the present stage of the research activity, there are possible failure of the DAKOTA application in SNAP if one MELCOR calculation fails.



DAKOTA uncertainty analysis for MELCOR code in a SNAP environment/architecture.

ENEA APPLICATIONS IN EU-MUSA PROJECT

- Currently if one calculation fails it prevents finalizing the UA application in SNAP:

o New Python directed job-stream capability in SNAP have been added;

o In the version 1.7 of the SNAP uncertainty plugin “the uncertainty quantification support in Python Directed streams was updated to support a specified number of "Replacement Samples" that are used to run additional tasks to replace those that fail to execute” (https://www.aptplot.com/snap/plugins/uncertainty/changes.jsp).

- The MELCOR/DAKOTA coupling through the Python Stream in the SNAP environment/architecture has been developed together with UNIPI in collaboration with ENEA.

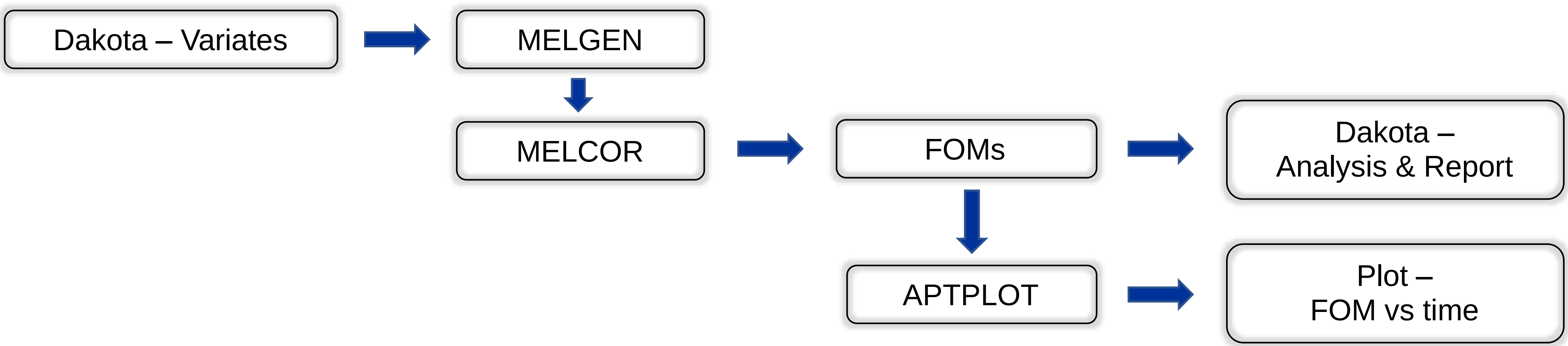
- Through this coupling approach, the management of failed code runs is possible by the last SNAP version (3.1.6).

- This MELCOR/DAKOTA coupling is managed through a python script, elaborate by SNAP, which permits to:

- Run the MELGEN/MELCOR runs with the different sets of input uncertain parameters, created by the DAKOTA uncertainty plug-in;
- Calculate the FOM values for each run;
- Plot the dispersion of the FOMs through the module “PyPost”, developed by AptPlot;
- Generate the UQ final report.

ENEA APPLICATIONS IN EU-MUSA PROJECT

MELCOR and DAKOTA coupling through SNAP:
PYTHON DIRECTED STREAM



The activity has been done in collaboration with University of Pisa.

ENEA APPLICATIONS IN EU-MUSA PROJECT

- Considering the current issues when using the MELCOR/DAKOTA coupling with the DAKOTA plug-in in SNAP for uncertainty analysis a new mixed approach through SNAP and MATLAP has been developed by UNIPI.
- New Python Directed job-stream feature has been added in SNAP.

SNAP

- Uncertainty Configuration
- Management of Code Runs

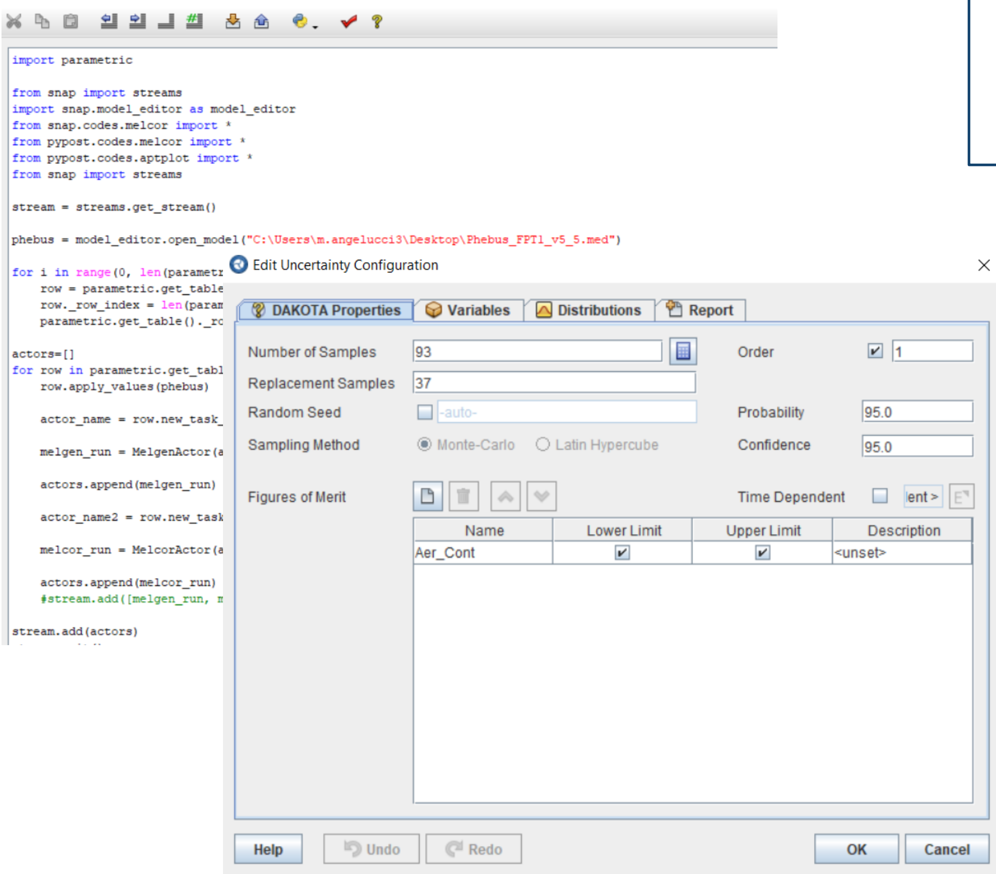
MATLAB

- Statistical Analysis
- Correlation Coefficients
- Plots & Brief Report

The analysis of the MELCOR and DAKOTA coupling through a mixed approach SNAP/MATLAB is under development along the EU-MUSA project by UNIPI

ENEA APPLICATIONS IN EU-MUSA PROJECT

SNAP



Use of DAKOTA plug-in
&
new Python Directed Stream feature

- Input Parameters
- Distributions
- Sampling
- N° of runs
- Parallel Computing

ENEA APPLICATIONS IN EU-MUSA PROJECT

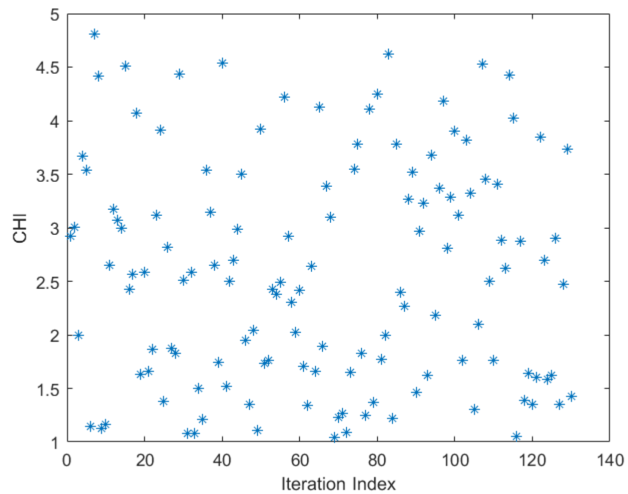
MATLAB

Parameters related data extracted from «parametric.py»

PARAMETERS_TABLE

	A	B	C	D	E	F	G	H
1	CHI	GAMMA	FSLIP	STICK	TURBDS	TKGOP	FTHERM	DELDIF
2	2.917758	1.384461	1.240084	0.902309	0.000832	0.00639	2.495339	5.41E-05
3	3.007051	3.043297	1.229796	0.803784	0.001171	0.054991	2.38043	0.000166
4	1.995144	3.10434	1.237184	0.937509	0.001079	0.02168	2.241081	6.77E-06
5	3.667595	1.012699	1.269932	0.930263	0.001013	0.0121	2.434387	0.000136
6	3.540712	4.087984	1.23822	0.863357	0.000803	0.009759	2.240758	0.00017
7	1.145789	3.387977	1.261731	0.979848	0.001112	0.006511	2.206781	0.000136
8	4.811028	3.050262	1.246044	0.569483	0.001169	0.00762	2.148951	0.000164
9	4.411897	1.136382	1.265689	0.814381	0.000853	0.035548	2.317295	8.17E-05
10	1.127445	1.637007	1.240215	0.703521	0.000979	0.010862	2.087603	0.000141
11	1.159056	3.560364	1.261965	0.676546	0.001031	0.050393	2.172414	0.000139
12	2.652567	1.934348	1.270717	0.608234	0.001236	0.021614	2.028366	4.34E-05
13	3.175211	3.486956	1.271715	0.565414	0.00122	0.015662	2.274189	5.42E-05
14	4.513611	3.403932	1.238271	0.907929	0.001017	0.007463	2.045325	2.51E-05
15	2.569383	1.949613	1.258938	0.981569	0.001164	0.050946	2.156422	1.82E-05
16	1.634272	3.943583	1.252372	0.859384	0.000884	0.056702	2.481325	8.18E-05
17	2.585649	2.690455	1.240403	0.783616	0.001136	0.006537	2.202507	1.03E-05

PARAMETER vs Iter.



ENEA APPLICATIONS IN EU-MUSA PROJECT

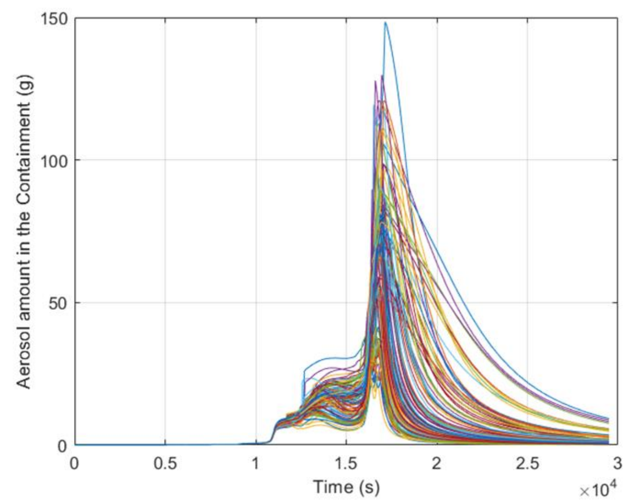
MATLAB

FOM values extracted from
«FOM.txt» (EDF)

FOM_TABLE

	Z	AA	AB	AC	AD	AE	AF	AG	AH
1	0	0	0	0	0.11224	0.113325	0.111495	0.10961	0.107784
2	0	0	0	0	0.113002	0.114099	0.112257	0.110359	0.10852
3	0	0	0	0	0.102463	0.103147	0.10129	0.099393	0.097558
4	0	0	0	0	0.115934	0.117092	0.115226	0.113302	0.111438
5	0	0	0	0	0.115756	0.116898	0.115022	0.113089	0.111215
6	0	0	0	0	0.091307	0.092003	0.090389	0.088726	0.08712
7	0	0	0	0	0.119434	0.120651	0.118741	0.116773	0.114865
8	0	0	0	0	0.118123	0.11932	0.117427	0.115476	0.113585
9	0	0	0	0	0.090862	0.091583	0.090008	0.088382	0.086814
10	0	0	0	0	0.091251	0.09195	0.090337	0.088675	0.087071
11	0	0	0	0	0.111234	0.112291	0.110469	0.108592	0.106774
12	0	0	0	0	0.114025	0.115131	0.113274	0.111361	0.109507
13	0	0	0	0	0.118389	0.119567	0.117656	0.11569	0.113783
14	0	0	0	0	0.109569	0.11056	0.108734	0.106855	0.105036
15	0	0	0	0	0.100213	0.10106	0.09934	0.097569	0.095856
16	0	0	0	0	0.109008	0.109925	0.108065	0.106157	0.10431
17	0	0	0	0	0.100797	0.101616	0.099871	0.098076	0.09634

DISPERSION

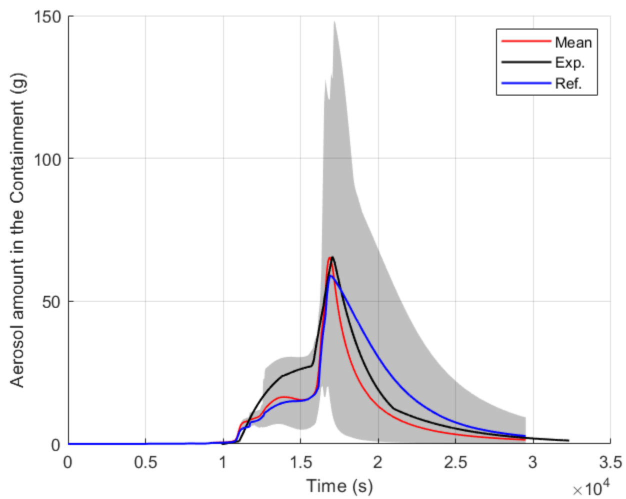


MATLAB

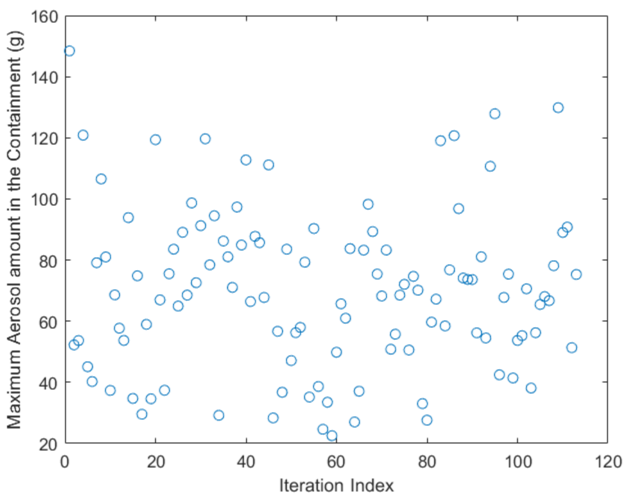
STATISTICAL ANALYSIS

(max, min, mean, median, standard deviation, skewness, kurtosis, CDF, PDF, …)

TIME-DEPENDENT



MAX_VALUE



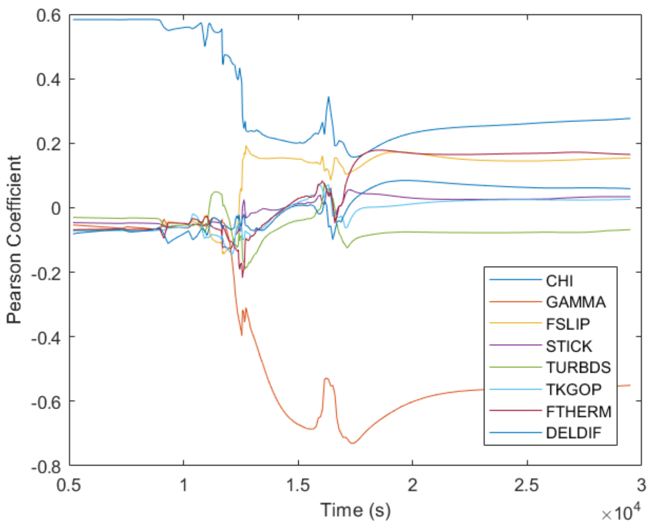
ENEA APPLICATIONS IN EU-MUSA PROJECT

MATLAB

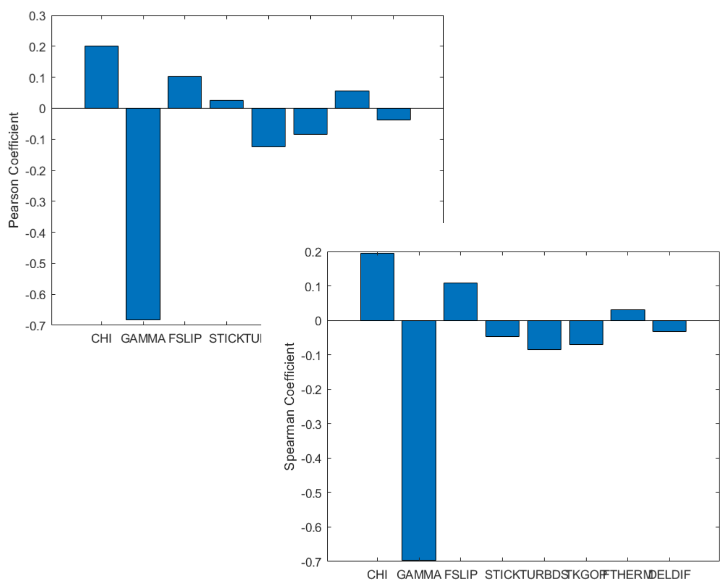
SENSITIVITY ANALYSIS

(Pearson & Spearman coefficients)

TIME-DEPENDENT



MAX_VALUE



MATLAB

BRIEF REPORT

Main information about
failed runs & statistical quantities

```
Brief Report

number of failed runs: 17

failed runs: 13 14 16 18 26 29 36 38 39 48 49 66 84 100 104 124 126

selected FOM: maximum Aerosol amount in the Containment (g)
maximum value: 148.4348
minimun value: 22.5524
mean value: 69.8765
median value: 68.5436
standard deviation: 26.015
quantile 0.95: 119.6295
quantile 0.05: 29.2779
skewness: 0.42559
kurtosis: 2.9877
```

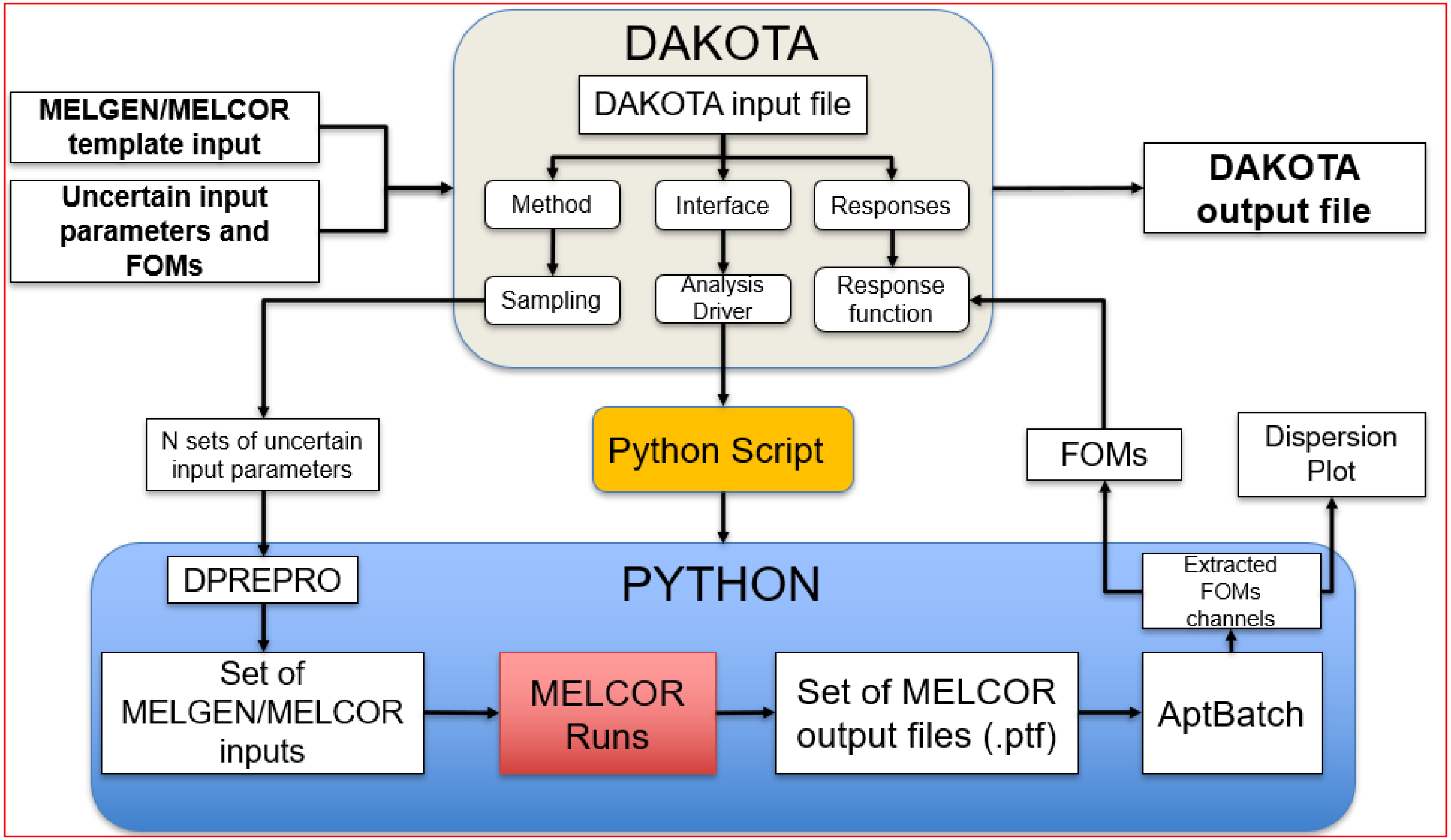
ENEA APPLICATIONS IN EU-MUSA PROJECT- Coupling MELCOR/DAKOTA with Python Script

- Implementation of the coupling MELCOR/DAKOTA with Python Script*.
- Coupling based on using the user interface DAKOTA GUI implemented by DAKOTA.
- Goal of this application is to resolve the issue of failed MELCOR runs, develop the uncertainty quantification report, and explore the full capability of DAKOTA stand alone.
- This approach is based on the Python scripting which permits to:
  - Run the MELCOR cases;
  - Extract the FOMs channels with AptBatch executable;
  - Plot the dispersion of FOMs channels;
  - Compute the statistical values of the FOMs (e.g. mean, median, upper and lower bounds, quantiles, ecc.) in a time-dependent approach;
  - Compute and plot the time-dependent Pearson and Spearman's coefficients;
  - Compute and plot the PDF and CDF of the FOMs.

- A dedicated presentation has been done during the EMUG22.

*A PhD student of the University of Palermo is part of this activity

ENEA APPLICATIONS IN EU-MUSA PROJECT- Coupling MELCOR/DAKOTA with Python Script



The analysis of the MELCOR and DAKOTA coupling through Python Scripts has been developed along the EU-MUSA project

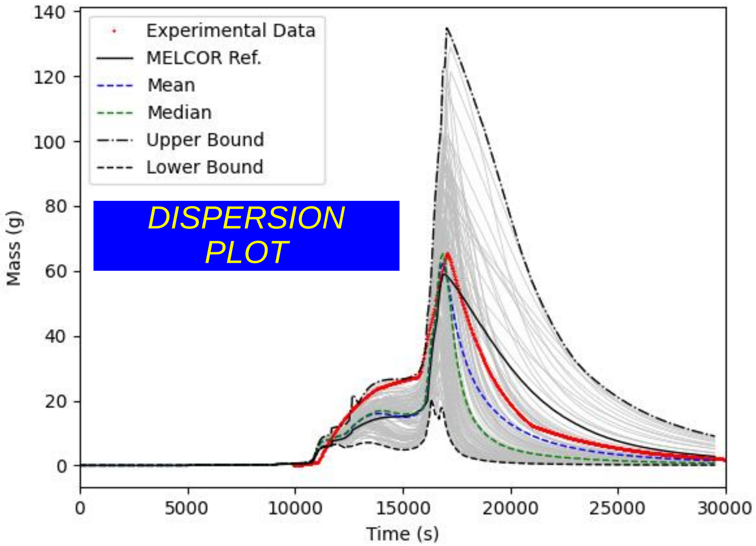
ENEA APPLICATIONS IN EU-MUSA PROJECT

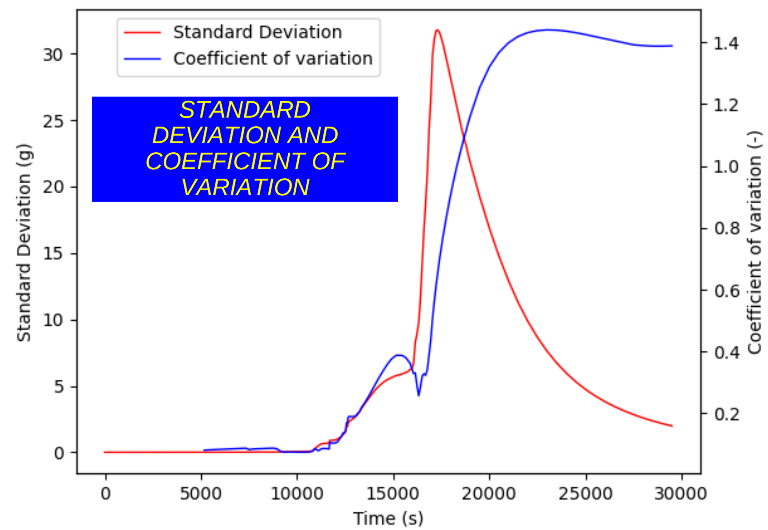
- The uncertain input parameters selected from the WP2 database for the uncertainty analysis currently are:
  - CHI (Aerosol dynamic shape factor)
  - GAMMA (Aerosol agglomeration shape factor)
  - FSLIP (Particle slip coefficient)
  - STICK (Particle sticking coefficient)
  - TURBDS (Turbolence dissipation rate)
  - TKGOP (Ratio of the thermal conductivity of the gas over that for the particle)
  - FTHERM (Thermal accomodation coefficient)
  - DELDIF (Diffusion boundary layer thickness).

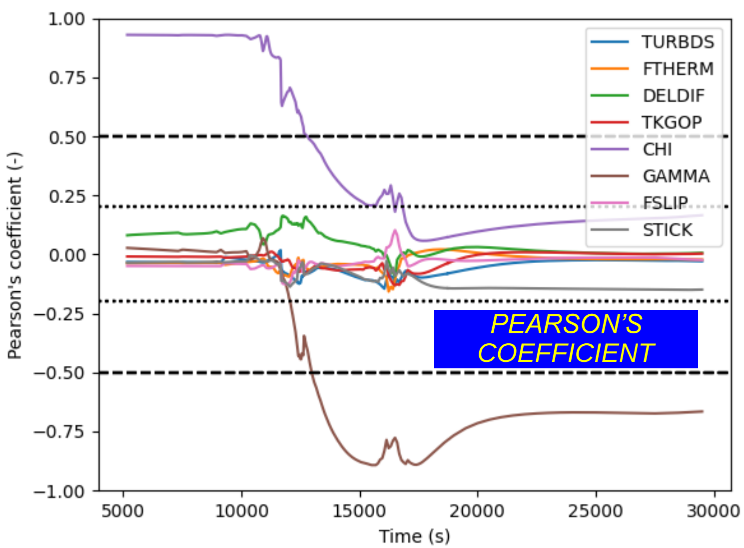
- At the current state of the activity, the preliminary FOM selected for the uncertainty analysis is the «Aerosol mass in suspension in the containment's atmosphere» and in particular the statistical analysis is conducted considering the maximum value of this one.

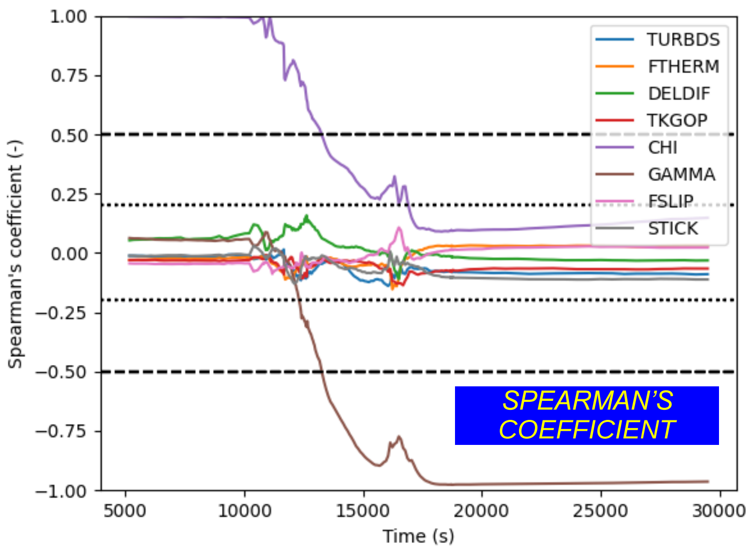
ENEA

ENEA APPLICATIONS IN EU-MUSA PROJECT





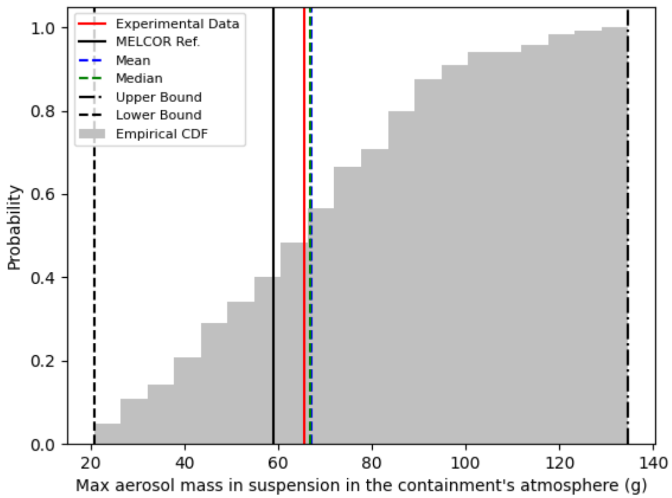




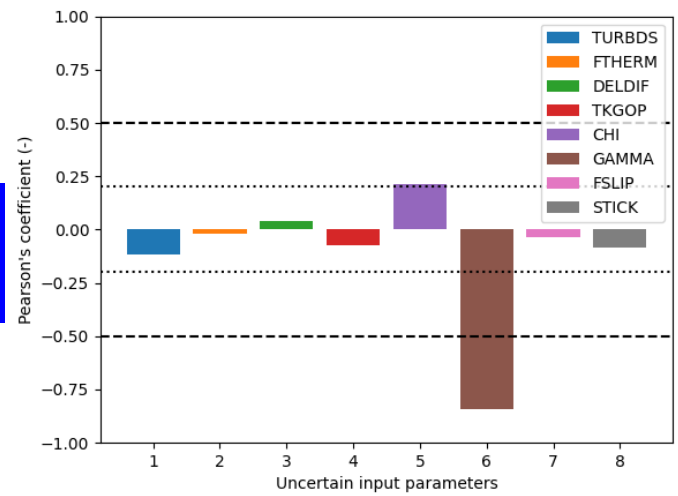
The activity has been done in collaboration with University of Palermo.

ENEA APPLICATIONS IN EU-MUSA PROJECT

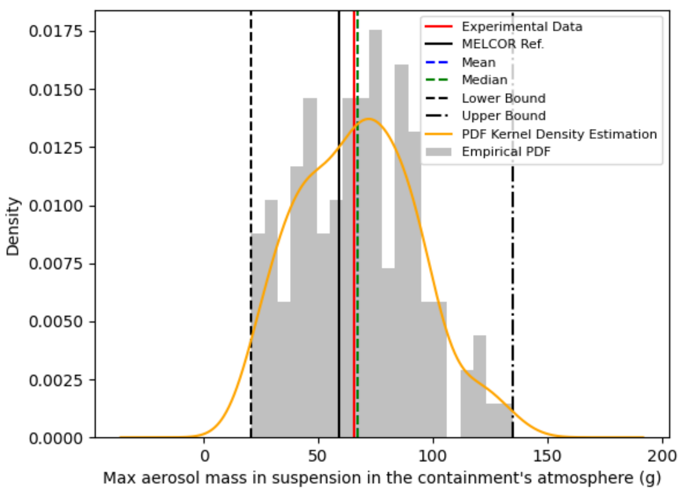
CDF AND PDF OF THE MAXIMUM OF THE FOM CONSIDERED AND PEARSON AND PEARSON COEFFICIENT



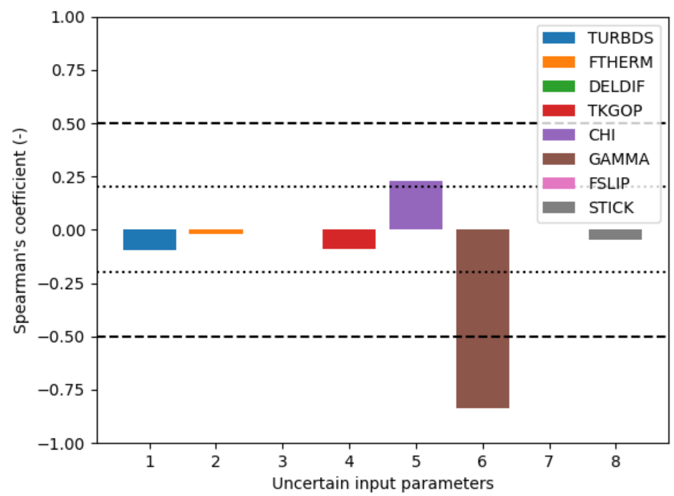
CDF OF THE MAXIMUM VALUE



PEARSON'S COEFFICIENT OF THE MAXIMUM VALUE



PDF OF THE MAXIMUM VALUE



SPEARMAN'S COEFFICIENT OF THE MAXIMUM VALUE

ENEA APPLICATIONS IN EU-MUSA PROJECT

- At the present stage of the research activity, it has been possible develop a reference case for the Phebus FPT1.

- The FOMs investigated are "Release of iodine from the test fuel bundle", "Release of Caesium from the test fuel bundle", "Caesium retention in the circuit", "Aerosol mass in suspension in the containment's atmosphere" and "Total deposited iodine in the containment".

- The probabilistic method to propagate input uncertainty by coupling MELCOR/DAKOTA in the SNAP environment/architecture has been chosen but some issues related to MELCOR runs failure do not allow the finalization of the uncertainty analysis.

- A MELCOR/DAKOTA coupling with Python has been developed. The uncertainty input parameters, selected from the WP2 have been added in the input model and the uncertainty application, assuming the "Aerosol mass in suspension in the containment's atmosphere" as FOM for the statistical analysis, has been developed.

- Challenges:
  - Implementation of Python Stream in SNAP to resolve the issues related to possible MELCOR runs fail.
  - Implementation on the cluster of MELCOR/DAKOTA coupling.

ENEA APPLICATIONS IN EU-MUSA PROJECT

[1] Schwarz, M., Clement, B., Jones, A.V., 2001. Applicability of Phebus FP results to severe accident safety evaluations and management measures. Nucl. Eng. Des. 209, 173–181.

[2] Dubourg, R., Faure-Geors H., Nicaise, G., Barrachin, M., 2005. Fission product release in the first two PHEBUS tests FPT0 and FPT1. Nuclear Engineering and Design 235, 2183–2208

[3] Darnowski, P., Włostowski, M., Stępień, M., Niewiński, G., 2020. Study of the material release during PHÉBUS FPT-1 bundle phase with MELCOR 2.2, Annals of Nuclear Energy 148.

[4] B. Clément et al., «Thematic network for a Phebus FPT1 international standard problem (THENPHEBISP)», Nuclear Engineering and Design 235, pp. 347-357 (2005).

[5] Bosland, L., Weber, G., Klein-Hessling, W., Girault, N. & Clement, B., 2012. Modeling and Interpretation of Iodine Behavior in PHEBUS FPT-1 Containment with ASTEC and COCOSYS Codes, Nuclear Technology, 177:1, 36-62.

ENEA APPLICATIONS IN EU-MUSA PROJECT (WP5)

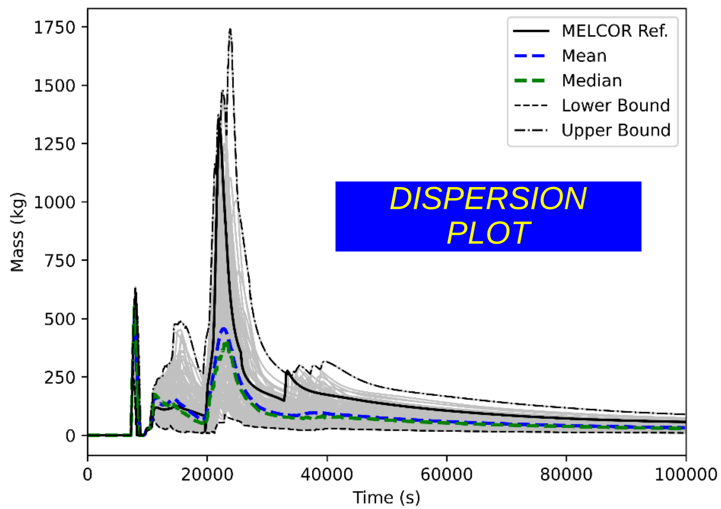
- Along the WP5, named UQAMRA (Uncertainty Quantification in Analysis and Management of Reactor Accidents), of the EU-MUSA Project, the Uncertainty Analysis (UA) of a Station Black-Out (SBO) in a generic PWR-900 MWe three-loops western type has been developed.

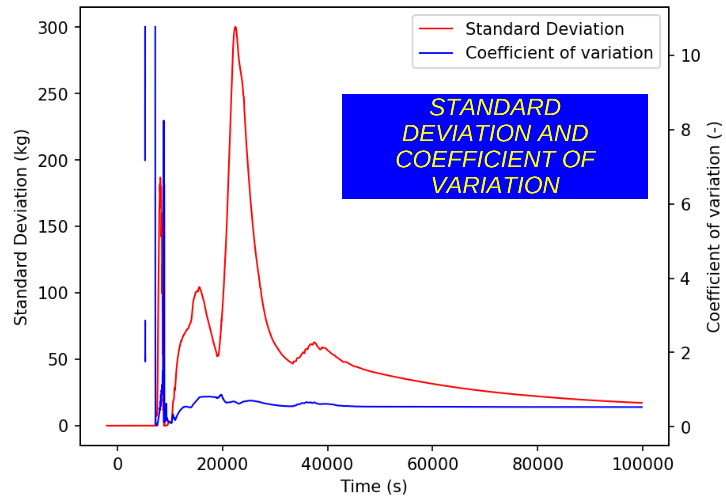
- To carry out the analysis, the MELCOR/DAKOTA coupling in a Python environment/architecture developed along the MUSA WP4 has been used.

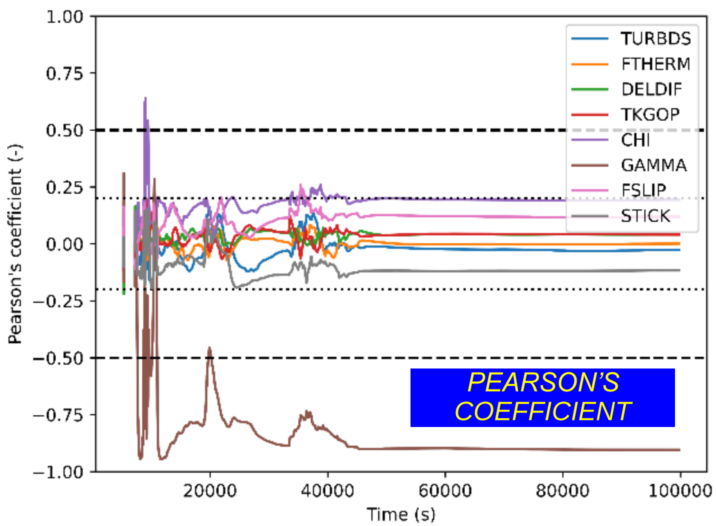
- The aerosol miscellaneous constants have been selected as uncertain input parameters and the aerosol suspended mass in the containment's atmosphere as Figure of Merit (FOM).

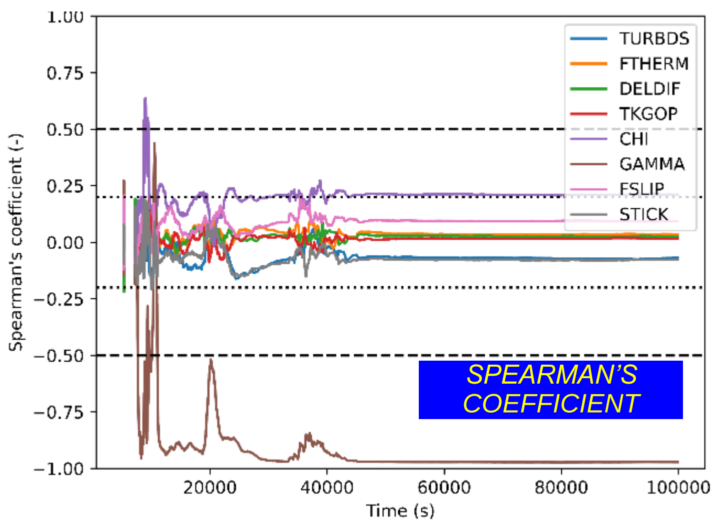
- To conduct the UA the probabilistic method to propagate input uncertainty has been used. Based on Wilks, in case only one FOM is investigated and for the two-sided tolerance interval, a minimum of 93 code runs is required for a probability and confidence level of 95%. To consider potential code runs failures, a total of 130 code runs have been performed. In this application the failed code runs have not been considered for the UA.

ENEA APPLICATIONS IN EU-MUSA PROJECT (WP5)



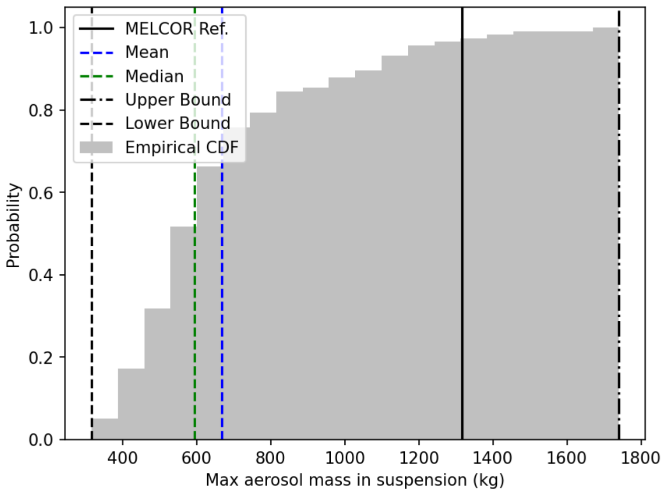




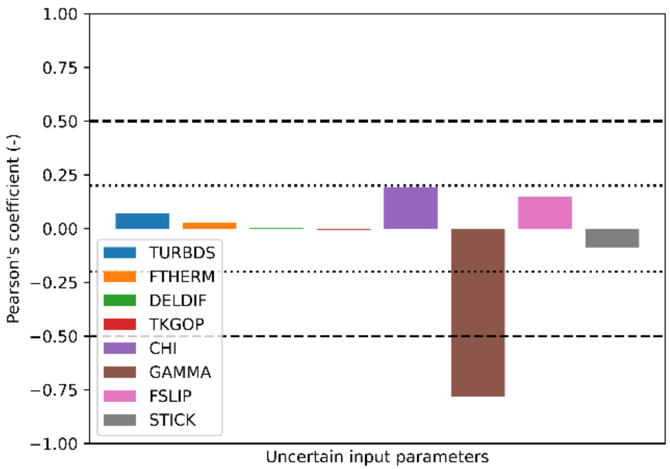


The activity has been done in collaboration with University of Palermo.

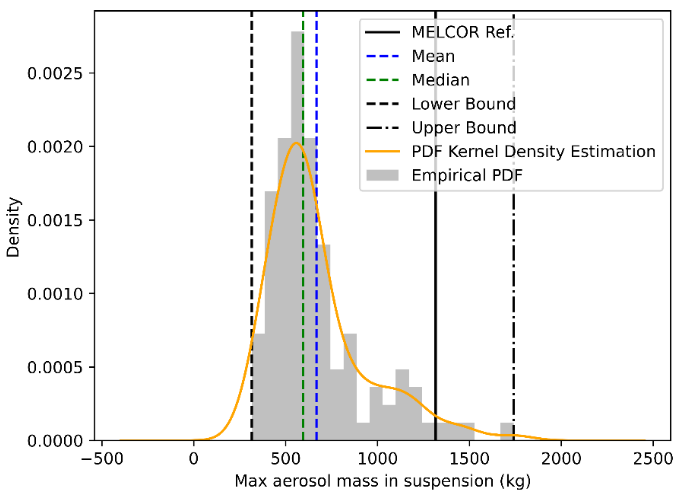
ENEA APPLICATIONS IN EU-MUSA PROJECT (WP5)



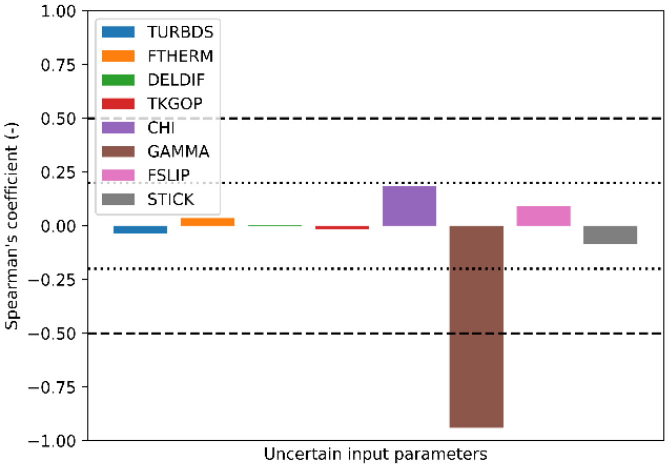
CDF OF THE MAXIMUM VALUE



PEARSON'S COEFFICIENT OF THE MAXIMUM VALUE



PDF OF THE MAXIMUM VALUE



SPEARMAN'S COEFFICIENT OF THE MAXIMUM VALUE

ENEA INVOLVMENT IN EU-MUSA PROJECT

This project has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 847441

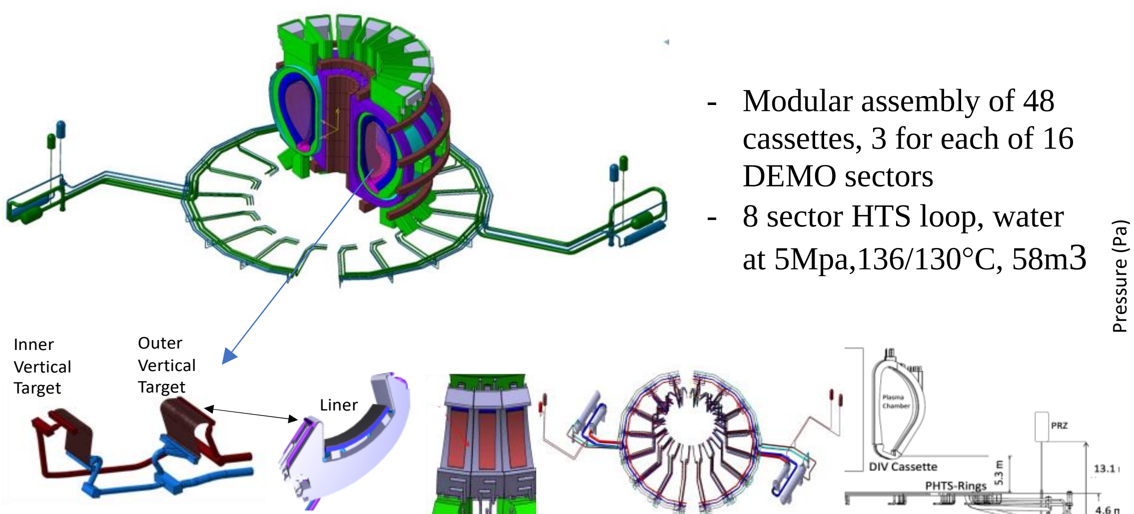
ENEA INVOLVMENT IN EUROFUSION - Accident analyses within EUROFusion project

ENEA INVOLVMENT IN EUROFUSION - Accident analyses within EUROFusion project:

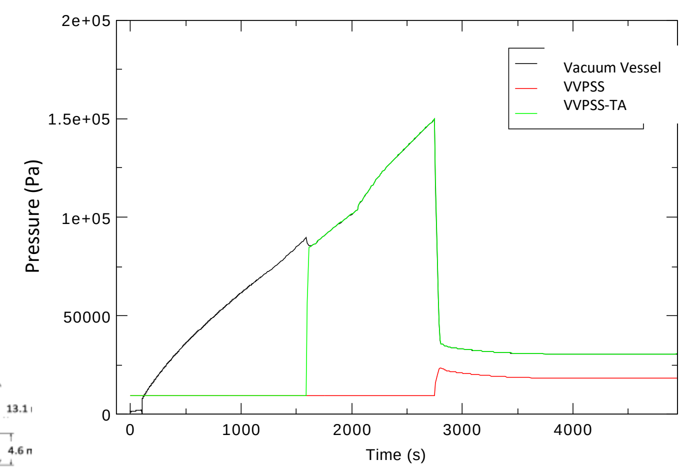
❑ DEMO Tokamak plant: Divertor system analyses

- Preliminary Safety Assessment of Conceptual design of European Nuclear Fusion DEMOnstrative Tokamak reactor plant
- Melcor v1.8.6 fusion, water coolant
- Design Basis Accident considered:
  - Postulated Initiating Event: a loss of coolant in the DV primary cooling circuit of the PFU inside the VacuumVessel (VV).
  - Demonstrate Vacuum Vessel pressurization within safety margins (<0.2 Mpa), considered safety provisions, e.g. Vacuum Vessel Pressure Suppression System (VVPSS).

- Modular assembly of 48 cassettes, 3 for each of 16 DEMO sectors
- 8 sector HTS loop, water at 5Mpa,136/130°C, 58m3

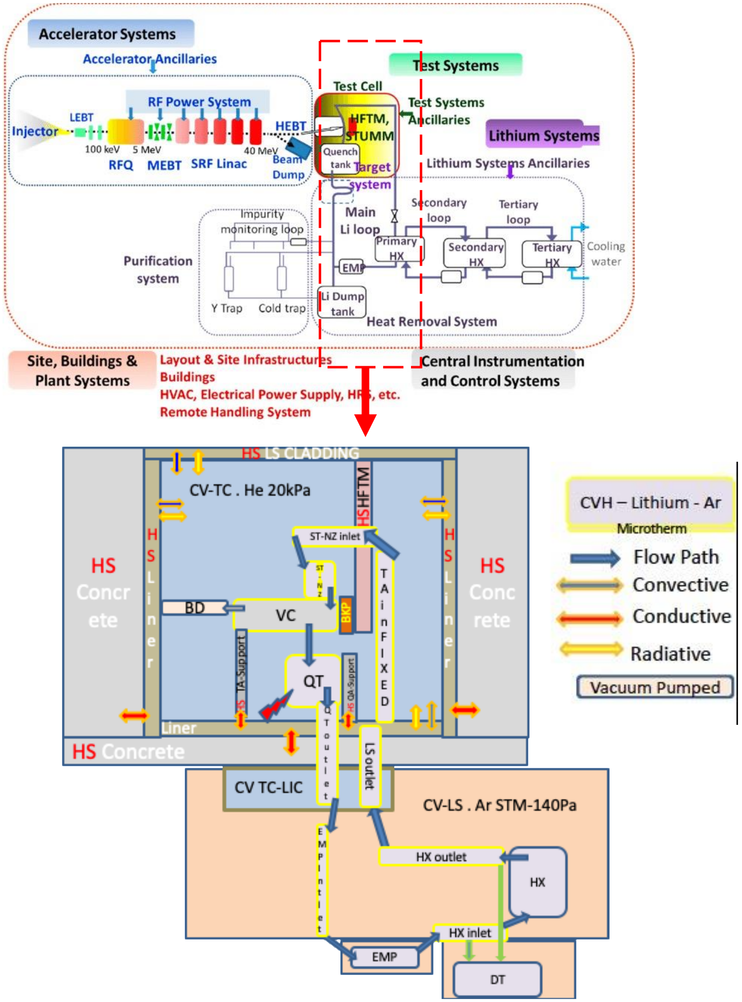


a) PFU module
b) Cassette module
c) Sector cassettes assembly
d) Divertor HTS layout
e) PFU/Cassette HTS elevations

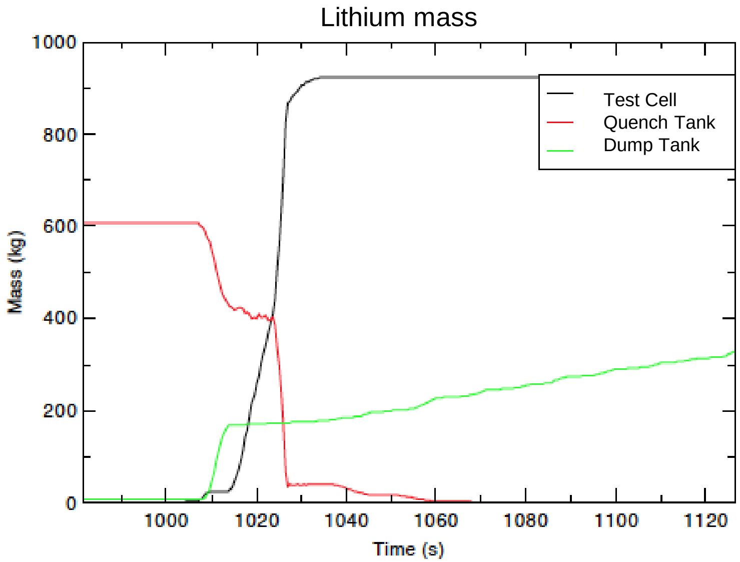


ENEA INVOLVMENT IN EUROFUSION - Accident analyses within EUROFusion project:

DONES neutron Irradiation facility



- DONES: Accelerated deuterons impacting a liquid lithium flowing film undergo striping reactions providing neutrons in fusion relevant energy spectrum for material testing purposes
- Melcor v1.8.6 fusion, lithium coolant
- Initiating Event: Loss of Lithium from primary loop due to large break in Quench Tank (QT) in Test Cell
- Purpose:
  - Assess released inventory of lithium
  - Impact on Test Cell containment atmosphere, water cooled floor liner



ANALYSES OF THE PHYSICAL MODELS NECESSARY TO BE IMPLEMENTED IN MELCOR_2.2 FOR FUSION REACTOR SAFETY ANALYSES AND THE CURRENT MODEL ALREADY IMPLEMENTED IN MELCOR FUSION

ANALYSES OF THE PHYSICAL MODELS NECESSARY TO BE IMPLEMENTED IN MELCOR_2.2 FOR FUSION REACTOR SAFETY ANALYSES AND THE CURRENT MODEL ALREADY IMPLEMENTED IN MELCOR FUSION

- During the last well organized EMUG meeting in Zagreb (25-27 April 2018), there was a very interesting discussion about the physical models necessary to be implemented in MELCOR_2.2 for fusion reactor safety analyses and the current model already implemented in MELCOR fusion.
- Since the session4 ("GEN IV and Fusion Applications") was chaired by ENEA, ENEA proposed an action that was agreed by all the Colleagues attending the meeting: ENEA will contact all the EMUG Partners to collect the information about physical models necessary to be implemented in MELCOR_2.2 for fusion reactor safety analyses and the current model already implemented in MELCOR fusion.
- ENEA has already contacted all the EMUG Partners to collect the information about physical models necessary to be implemented in MELCOR_2.2 for fusion reactor safety analyses and the current models already implemented in MELCOR fusion.
- These are the information requested:
  - Description of phenomenon of interest;
  - Safety relevance of the phenomenon for fusion reactor;
  - Rank of importance (1: low; 2: medium; 3: High) ==> priority for code development;
  - If models to characterize the phenomenon have been already implemented in MELCOR fusion and the related version.
- The final report related to this activity is in a draft form for comments and will be soon finalized and distributed.
- The activity has been developed in collaboration with BELV, CCFE, CIEMAT, KIT, CVR, Jacbsen Analytics, JSI, POLITO, SANDIA, UNIROMA1, UNIPA, UNIPI.

ANALYSES OF THE PHYSICAL MODELS NECESSARY TO BE IMPLEMENTED IN MELCOR_2.2 FOR FUSION REACTOR SAFETY ANALYSES AND THE CURRENT MODEL ALREADY IMPLEMENTED IN MELCOR FUSION

ISSUE N	ISSUE DESCRIPTION	Priority	Complexity of implementation	MELCOR_FUSION REFERENCES
1	Introduce additional working fluids with multiphase capabilities	3		[2][3]
2	Implementation of the possibility to use different fluids in different circuits at the same time during the calculation	3		-
3	Introduce models for chemical reactions in the case of different working fluids	2		[6][7][13]
4	Model steam oxidation of the Plasma-Facing-Component (PFC)	2		[2][4]
5	Model air oxidation of the Plasma-Facing-Component (PFC)	2		[4]
6	Introduce models for aerosols turbolent and inertial deposition	2		[2]
7	Introduce models for aerosols deposition with different carrying gas and mixtures	2		[2]
8	Introduce aerosol resuspension model	2		[9]
9	Extend the deposition and resuspension modelling to take into account remnant magnetization effects	1		
10	Introduce models for aerosols transport in multifluid (multi-working fluid) simulation.	2		-
11	Implementation of specific heat transfer correlations for simulating He as working fluid in the geometry of interest.	2		-
12	Standard Scrubber model in FL Package for Helium.	1		-
13	Introduce dissolved NCG species within working fluids	2		-
14	Implement magnetic pump modelling (for design) and features (e.g. coast-down, etc)	1		-
15	Include MHD effects on heat transfer correlation and pressure drop evaluation (for design)	1		-
16	Extend the water properties below triple point temperature	2		[2]
17	Air condensation onto cryogenic structures	2		[4][14]
18	Helium condensation onto cryogenic structures	2		[2]
19	Allow low temperature operations (>3K) and cryogen working fluids	2		
20	Extend material physical properties to cryogenic range	3		
21	Enclosure radiant heat transfer	2		[2]
22	A common release MELCOR 2.x incl. fusion features	1		
23	Hydrogen tritium oxide transport			
24	Tritium transport			
25	Dust and Hydrogen explosion			

ANALYSES OF THE PHYSICAL MODELS NECESSARY TO BE IMPLEMENTED IN MELCOR_2.2 FOR FUSION REACTOR SAFETY ANALYSES AND THE CURRENT MODEL ALREADY IMPLEMENTED IN MELCOR FUSION

- Within the European MELCOR User Group (EMUG), organized in the framework of USNRC Cooperative Severe Accident Research Program (CSARP), an activity on the evaluation of the applicability of MELCOR 2.2 for fusion safety analyses has been launched and it has been coordinated by ENEA.

- The aim of the activity was to identify the physical models to be implemented in MELCOR 2.2 necessary for fusion safety analyses, and to check if those models are already available in MELCOR 1.8.6 for fusion version, developed by Idaho National Laboratory.

- From this activity, a list of modeling needs, emerged from the safety analyses of fusion-related installations, have been identified and described.

- A paper has been presented at the SOFT 2022 conference. The identified modeling needs are discussed, together with the current status of the MELCOR code development. The ultimate goal is to have in the near future a single integrated MELCOR release capable to treat both fission and fusion applications.

ENEA INVOLVMENT IN ITALIAN SAFETY ASSESSMENT ACTIVITY

ENEA INVOLVMENT IN ITALIAN SAFETY ASSESSMENT ACTIVITY

- The ENEA activities related to the MELCOR code are oriented to the simulation and evaluation of severe accident evolutions and source term for "safety assessment", mainly focusing on the characteristics of NPPs located at the Italian border.

- The activity is the basis for the development of a source term database to be used as an input for the MACCS code available in ENEA .

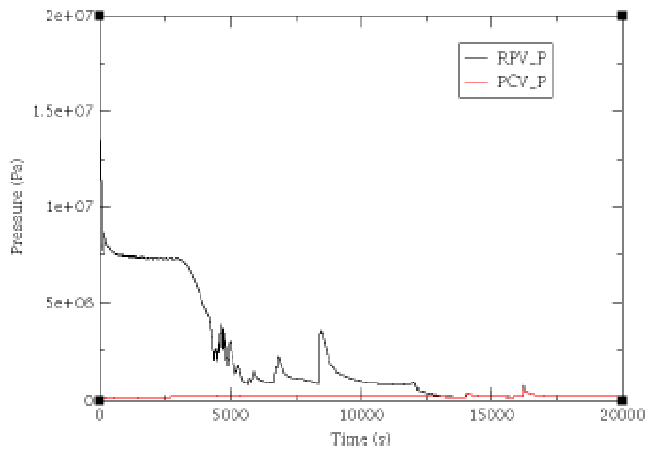
  - In-vessel analysis with the MELCOR 2.1 of three unmitigated LBLOCA severe accidents in a generic PWR of 900 MWe, caused by three distinct initiator events: a) Double-ended rupture of the cold leg of Loop 1; b) Double-ended rupture of the hot leg of Loop 1; c) Double-ended rupture of the surge line. The activity has been conducted in ENEA in the framework of a Master Degree thesis with the University of Bologna.

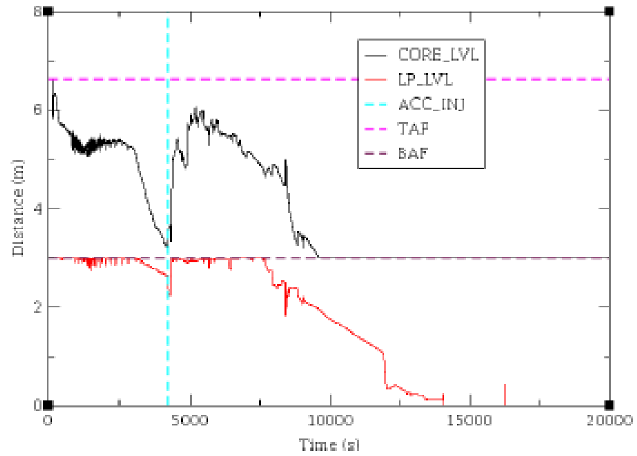
  - A short term Station Blackout (SBO) accident leading to a loss of the ultimate heat sink, and the possible thermal induced SGTR, with consequent evaluation of the source term has been calculated by MELCOR code.

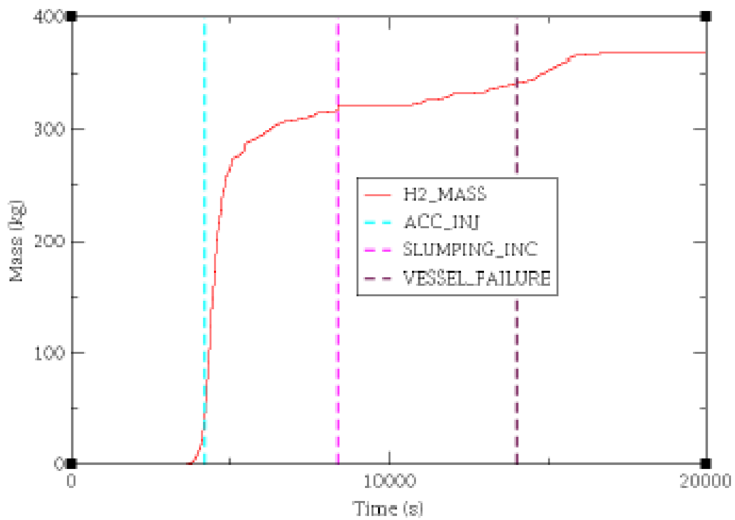
  - A 2 inch unmitigated SBLOCA has been calculated by MELCOR code:
    - The effect of the discharge coefficient at the break in the calculated results will be investigated.
    - The results could be compared with analogous ASTEC and MAAP results (independent user cross walk activity) done by JRC.

  - LFW and LBLOCA transient with MELCOR code have been performed by Sapienza University of Rome.

ITALIAN SAFETY ASSESSMENT ACTIVITY - 2 INCH UNMITIGATED SBLOCA







RELEVANT PHENOMENOLOGYCAL ASPECTS	MELCOR
SGs (1,2,3) Cycling Inception (s)	48, 50,50
Two Phase Inception in the HL 1,2,3 (s)	70,88,88
H2 Start (s)	2000
TCL 1300K (s)	3570*
TCL 1855K (s)	3870*
Start of Accumulators 1,2,3 Discharge (s)	4200
Upper Core Part Failure Inception (s)	4930
Massive Slumping Inception (s)	8390
Vessel Failure (s)	14000

* Radial ring 1 and axial level 16

ENEA INVOLVMENT IN ASCOM PROJECT

- The aim of this activity is to study the main phenomena during a postulate unmitigated 2-inch Cold Leg LOCA accident in a generic PWR 900 MWe three loops, especially as regards the thermal-hydraulic phases of the transient core degradation phenomena, the oxidation and hydrogen production processes.
- This transient is evaluated by the SA code ASTEC and compared with the results obtained with the SA code MELCOR.
- The transient selected for this analysis is initiated by a 2-inch break in the Cold Leg (CL) of the loop 1, where the Pressurizer (PRZ) is located. It is assumed that among the safety systems only the passive accumulators are in operation.
- The break event is coupled with:
  - Loss of offsite Alternating Current (AC) power;
  - Failure of all the diesel generators.

  This determines:
  - Primary pressure control systems (heaters and PRZ spry),
  - Chemical and Volume Control System (CVCS),
  - Reactor Coolant Pump (RCP) seal injection,
  - Active safety injection systems (High Pressure Injection System, HPIS, and Low Pressure Injecion System, LPIS),
  - Motor-driven Auxiliary Feedwater (MDAFW) system,

  are unavailable.

ENEA INVOLVMENT IN ASCOM PROJECT

- The following hypotheses are also considered:
  - Reactor SCRAM and Steam Generators (SGs) isolation at the Start Of the Transient (SOT);
  - Independent failure of the Turbine Driven Auxiliary Feedwater (TDAFW) pump (no AFW available);
  - No primary boundary structures thermal induced degradation phenomena
  - Primary and secondary Steam Relief Valves (SRVs) availability throughout the accident evolution.
  - Safety Valves of Pressure Compensator (SEBIM) manually stuck open when the core exit temperature reaches 650 °C.

- The transient has been analyzed until the Lower Head (LH) failure.

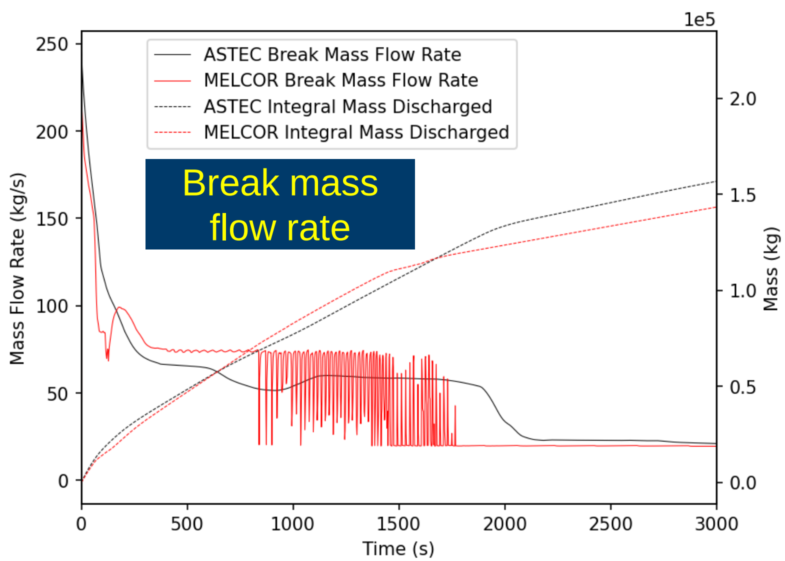
The activity has been done in collaboration with University of Palermo and University of Bologna.

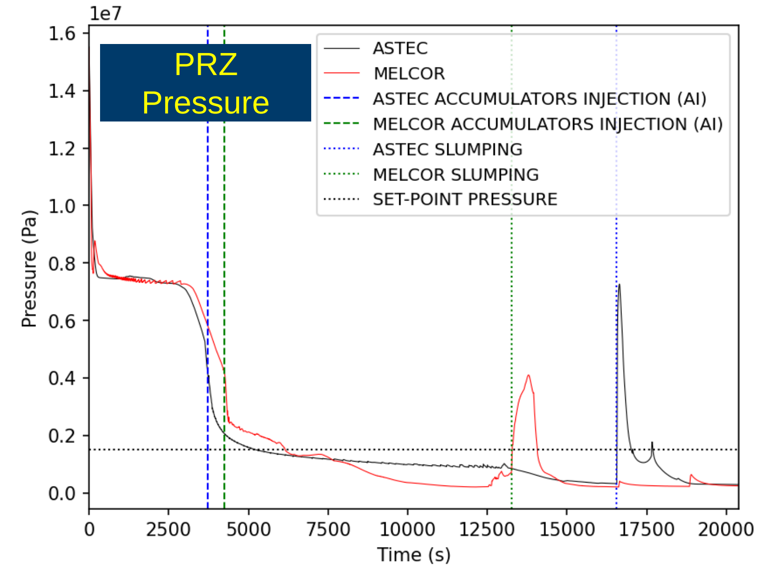
ENEA

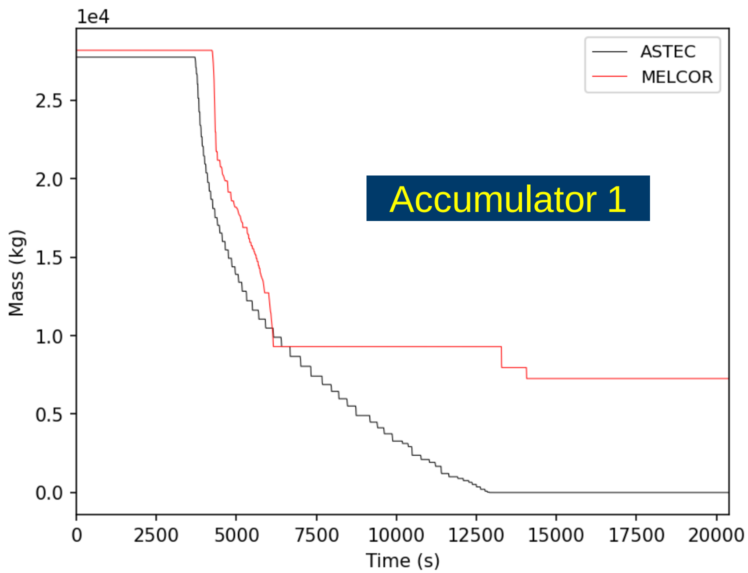
ENEA INVOLVMENT IN ASCOM PROJECT

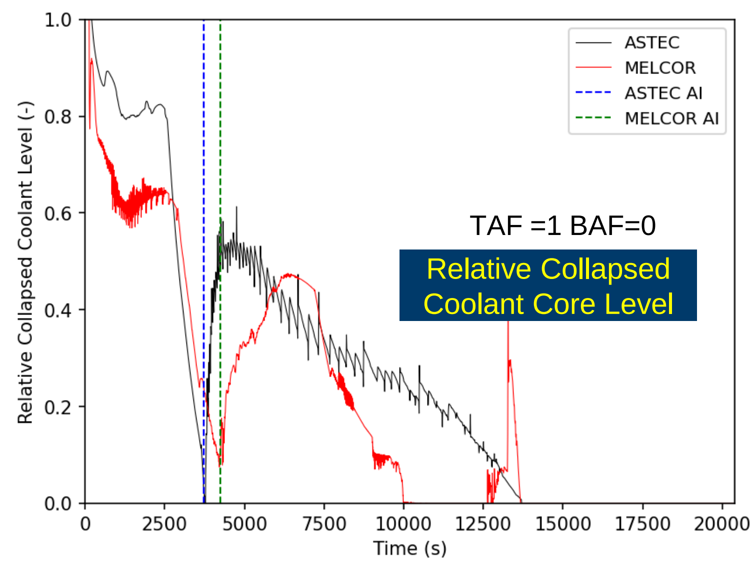
Relevant Phenomenological Aspects [s]	ASTEC	MELCOR	DISCR [%]
SOT, SCRAM, SGs isolation	0	0	0
SG 1,2,3 cycling inception	38,30,30	30,25,25	21.05,16.67,16.67
Core TAF uncovered	204	126	38.24
H_2 generation start	2923	3336	14.13
T > 1300 K (before accumulators injection)	-	3505	-
T > 1855 K (before accumulators injection)	-	3798	
Start of accumulators 1,2,3 discharge	3713,3713,3713	4236,4238,4238	14.09,14.14,14.14
Core BAF uncover	3773	-	-
T > 1300 K (after accumulators injection)	7208	8086	12.18
T > 1855 K (after accumulators injection)	7528	8707	15.66
Core BAF uncover	13702	10006	26.97
Slumping	16564	13253	19.99
LH failure	20394	20160	1.15

ENEA INVOLVMENT IN ASCOM PROJECT

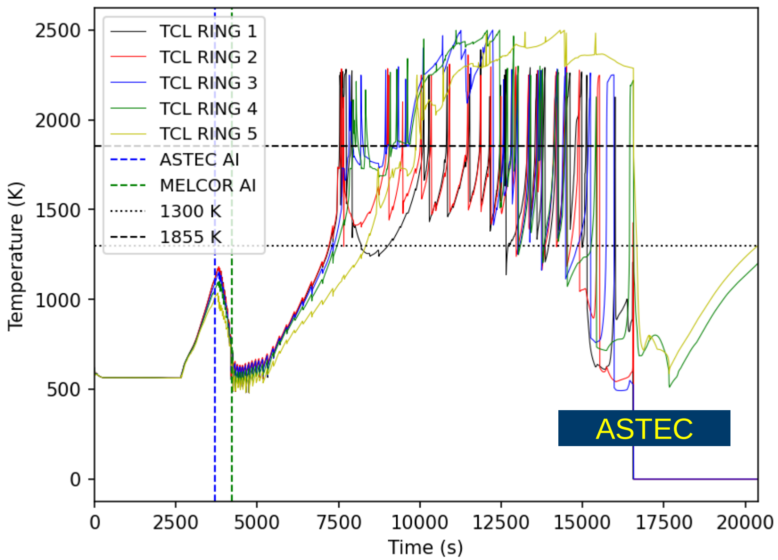


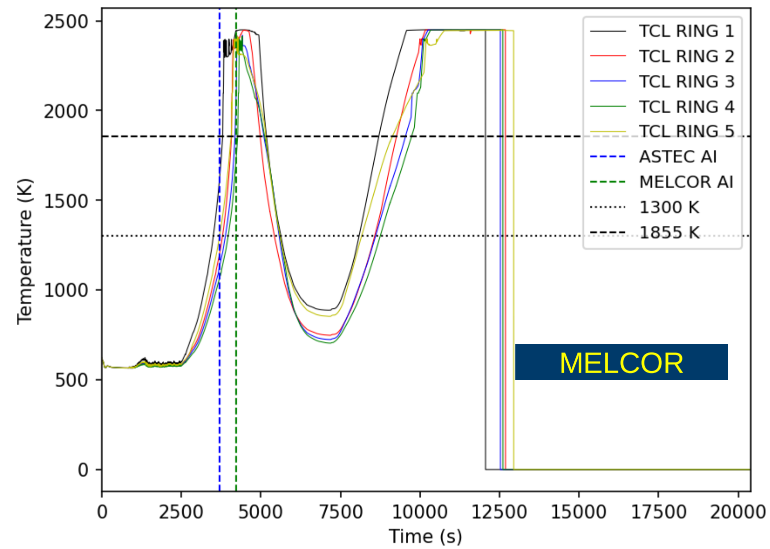


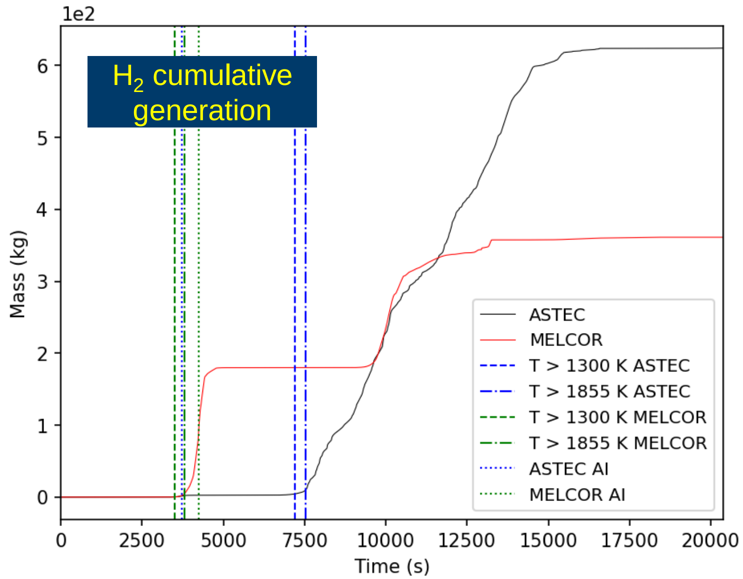


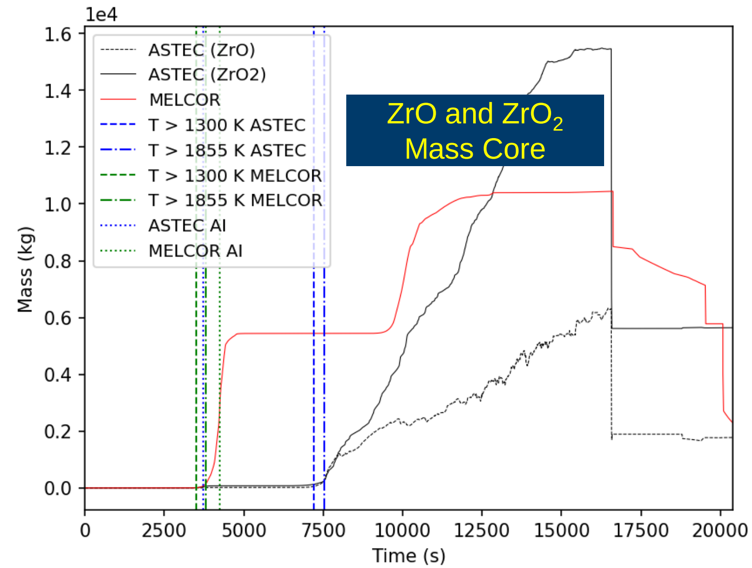


ENEA INVOLVMENT IN ASCOM PROJECT

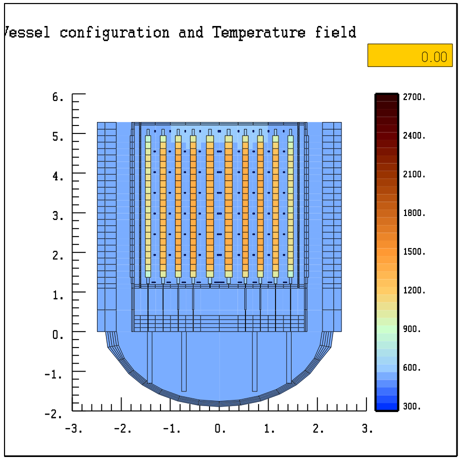


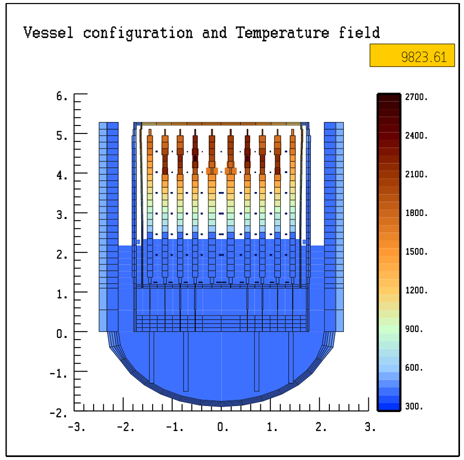


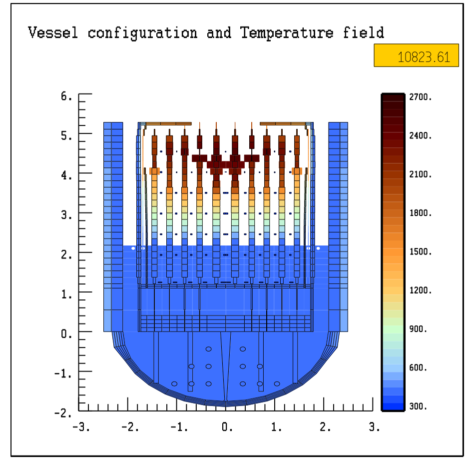


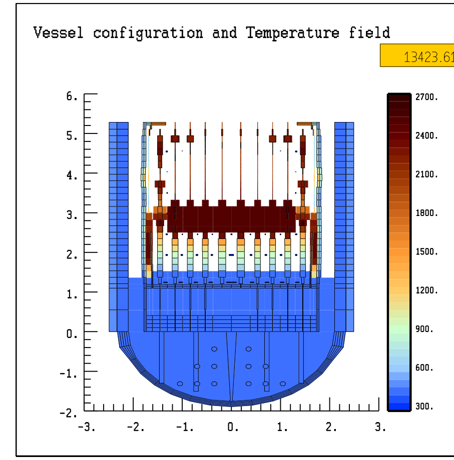


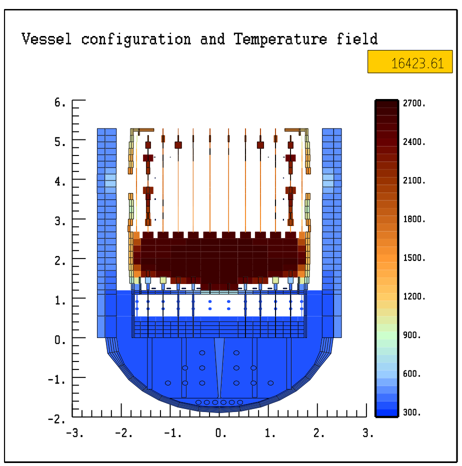
TRANSIENT ANALYSES: ASTEC CORE DEGRADATION ANIMATIONS

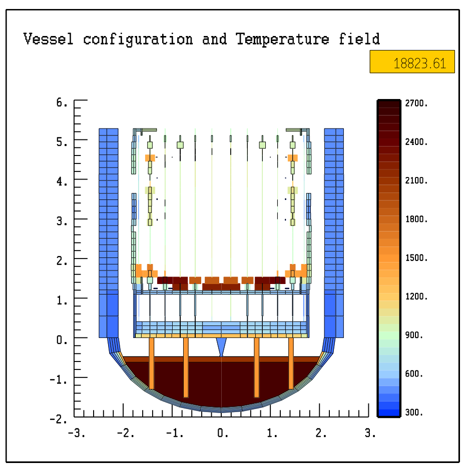


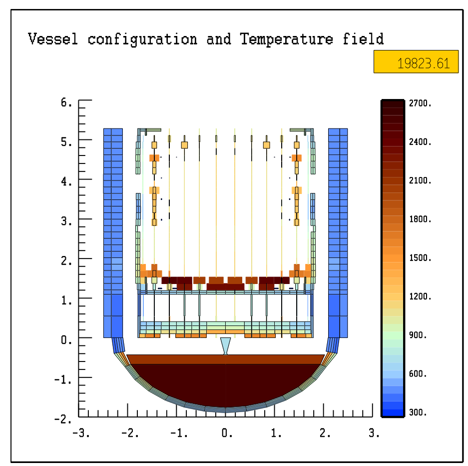


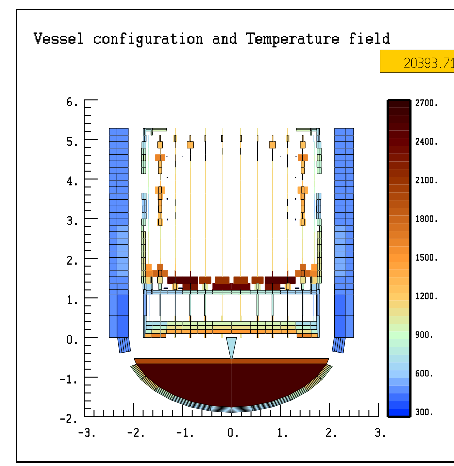




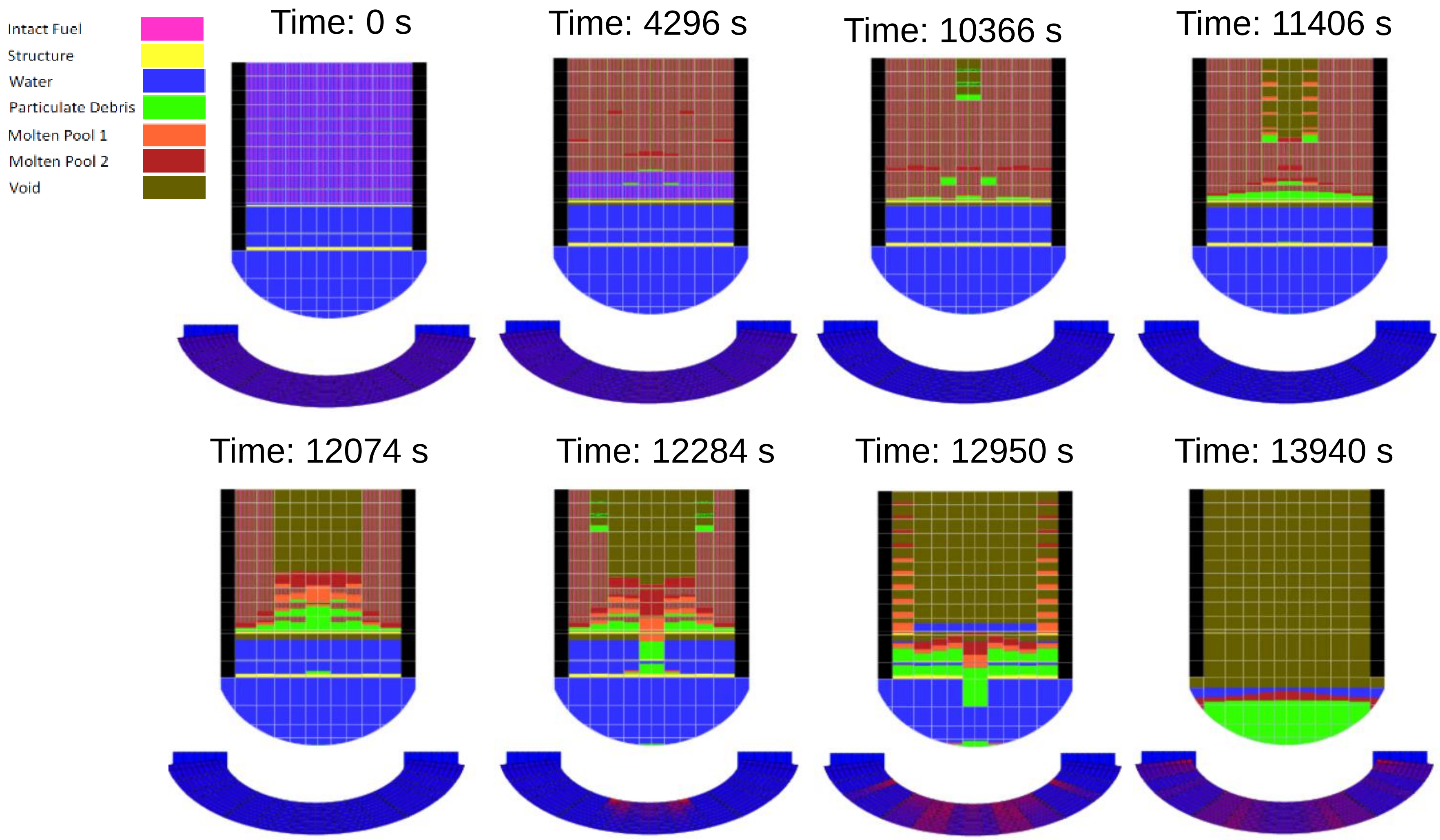








TRANSIENT ANALYSES: MELCOR CORE DEGRADATION ANIMATIONS



ENEA INVOLVMENT IN ASCOM PROJECT

- Similar thermal-hydraulic behaviour from the qualitative point of view between the two calculations even some quantitative differences are related to the break flow rate and the accumulator behaviour.
- Difference in terms of cladding temperature and the consequent H_2 generation before and after the accumulator injection.
- Even though MELCOR predict a faster degradation than ASTEC (also related to the different accumulator behaviour) and a consequent earlier corium relocation to the LP, the different retention time of the corium in the LP allow to have a discrepancy of about 1% for the LH failure, which occurs first in MELCOR than in ASTEC.
- The observed discrepancies underline some modelling differences between the two codes and details studies are in progress to characterize them and the user effect in view of uncertainty estimation.
- In relation to the user effect, in the case of MELCOR code the effect of the accumulator modelling is under investigation.

- *A paper has been published:*

G Agnello et al 2022 J. Phys.: Conf. Ser. 2177 012024 doi:10.1088/1742-6596/2177/1/012024

ACKNOWLEDGEMENTS

- Special thanks to USNRC, IRSN and Sandia National Laboratories for their comments during the preparation of the activities

SENSITIVITY ANALYSIS ON ACCUMULATORS MODELLING WITH MELCOR CODE

SENSITIVITY ANALYSIS ON ACCUMULATORS MODELLING WITH MELCOR CODE

- To give some insights about the code results discrepancies attributed to the different modelling approaches on the main SA phenomena, a sensitivity analysis on the accumulators modelling on the SA code MELCOR 2.2 has been carried out.
- The sensitivity analysis, presented in this work, has been carried out considering a generic PWR-900 MWe three-loops western type and an unmitigated Small Break Loss Of Coolant Accident (SBLOCA) as postulated SA scenario.
- The in-vessel phase has been analyzed and the effect of the accumulator modelling on the main thermal-hydraulic and core degradation phenomena have been investigated.
- The sensitivity analysis has been carried out considering two different accumulators modelling approaches:
    - In the first one, the accumulators have been modelled using Control Volume Hydrodynamics (CVH) Control Volumes (CVs).
    - In the second modelling approach, the accumulators have been modelled with the dedicated MELCOR ACC Package. A sensitivity analysis considering the isothermal and the adiabatic expansions. Furthermore, in order to investigate the effects of intermediate polytropic transformations between the isothermal and adiabatic ones, three additional cases, characterized by expansion coefficients of 1.1, 1.2 and 1.3 have been considered.

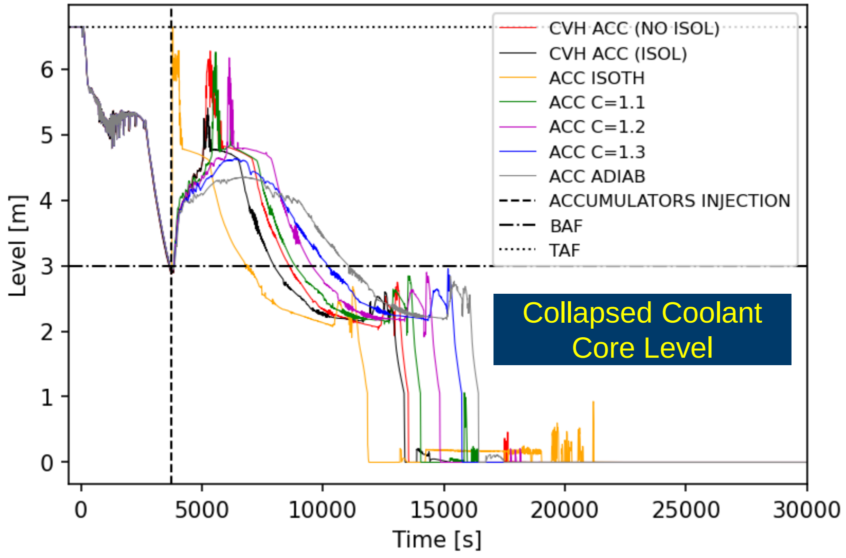
The activity has been done in collaboration with University of Palermo.

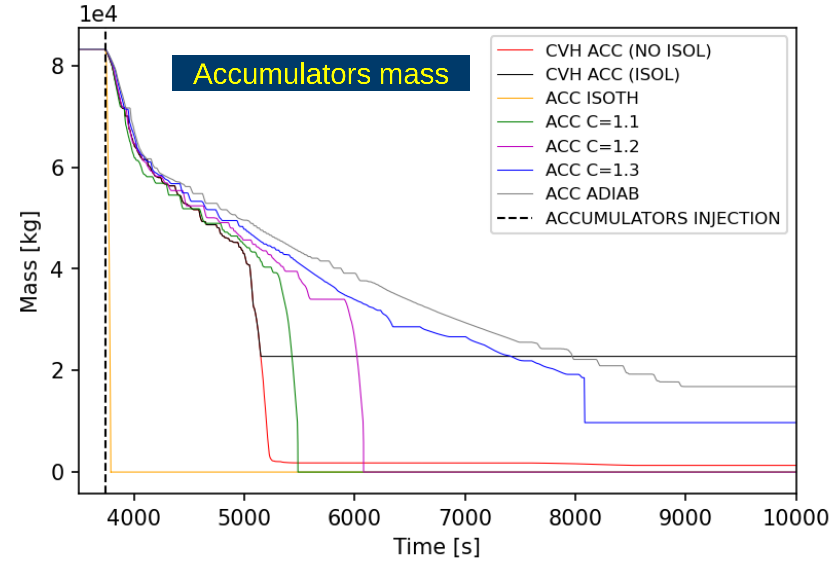
SENSITIVITY ANALYSIS ON ACCUMULATORS MODELLING WITH MELCOR CODE

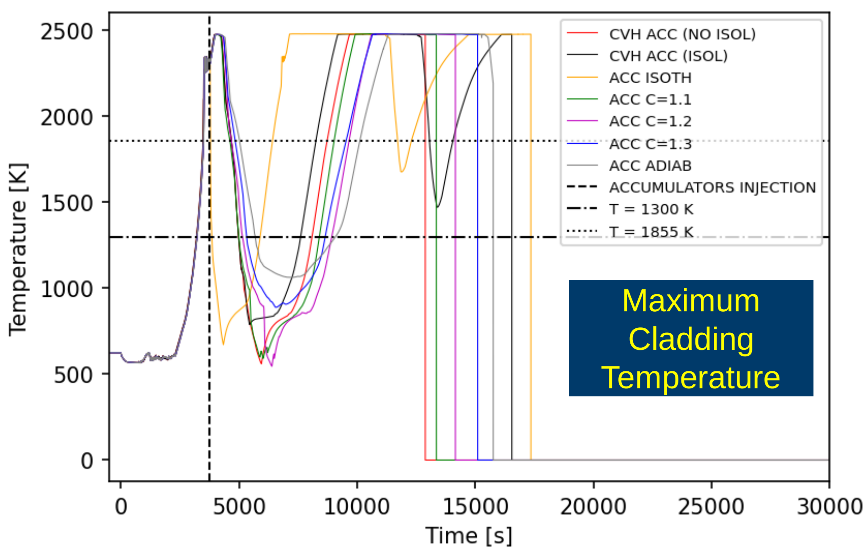
Relevant Phenomenological events [s]	CVH ACC		ACC				
	NO ISOL	ISOL	ISOTH	C = 1.1	C = 1.2	C = 1.3	ADIAB
SOT, SCRAM, SGs isolation	0	0	0	0	0	0	0
SG 1,2,3 cycling inception	25,20,20	25,20,20	25,20,20	25,20,20	25,20,20	25,20,20	25,20,20
Core TAF uncover	115	115	115	115	115	115	115
H_2 generation start	3065	3065	3065	3065	3065	3065	3065
T > 1300 K (before accumulators injection)	3205	3205	3205	3205	3205	3205	3205
T > 1855 K (before accumulators injection)	3480	3480	3480	3480	3480	3480	3480
Core BAF uncover (before accumulators injection)	3687	3687	3687	3687	3687	3687	3687
Start of accumulators discharge	3740	3740	3740	3740	3740	3740	3740
Stop of accumulators discharge	5350	5150	3780	5500	6050	8000	9000
T > 1300 K (after accumulators injection)	8095	7615	5895	8405	8925	8690	9090
T > 1855 K (after accumulators injection)	8725	8250	6415	9025	9655	9525	10110
Core BAF uncover (after accumulators injection)	9835	9500	7490	10174	10670	11500	12225
Core slumping	12980	12530	11200	13490	14270	15150	15780
LH failure	17486	18170	21115	18285	18900	19900	20840

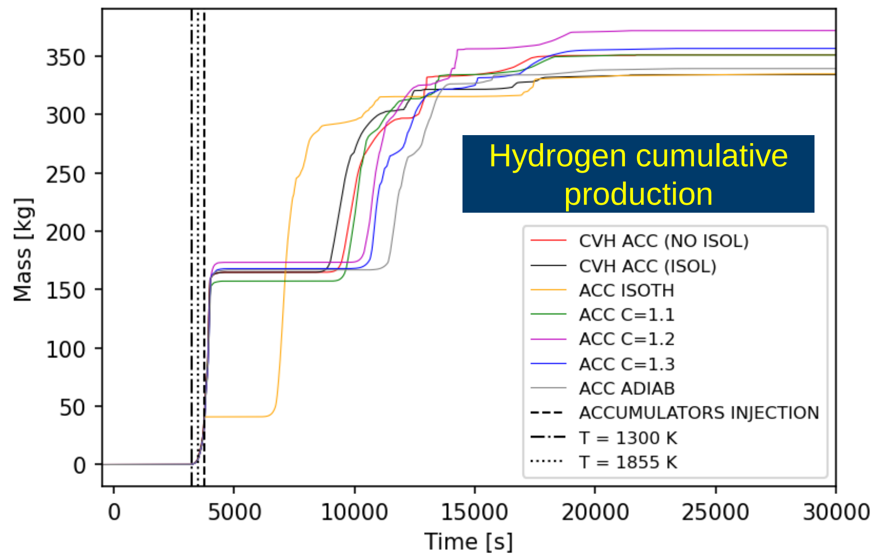
ENEA

SENSITIVITY ANALYSIS ON ACCUMULATORS MODELLING WITH MELCOR CODE









SENSITIVITY ANALYSIS ON ACCUMULATORS MODELLING WITH MELCOR CODE

- After the accumulators discharge, occurring at 3740 s after the SOT, all the model considered presents a similar trend in terms of water discharged from a qualitative point of view.
    - An exception is represented by the ACC ISOTH model predicting a fast discharge of all the water inventory in the PCS of the reactor.

- After 5000 s, the models considered presents a different behavior in terms of total mass discharge and mass flow rate influencing the core reflooding, cladding temperature and hydrogen rate production.
- The slower accumulator water injection in the case of ACC ADIAB determines a delay of the total core damage and the core slumping.
- The core relocated in the LP first in case of ACC ISOTH the LH failure occurs first in case of CVH ACC without accumulators isolation, probably due to the major quantity of steam available for oxidation reaction, determining a major core damaging. In case of ACC ISOTH the LH failure occurs at about 21115 s after the SOT and a less quantity of corium material relocated along the slumping is underlined.
- A paper has been presented at ERMSAR22.

Development of a Design 2 MELCOR input-deck in the frame of SASPAM-SA

HORIZON EURATOM SASPAM-SA PROJECT

Key Object and Outcomes

SAFETY ANALYSIS OF SMR WITH PASSIVE MITIGATION STRATEGIES - SEVERE ACCIDENT

- SASPAM-SA (Safety Analysis of SMR with Passive Mitigation strategies - Severe Accident) project proposal has been funded in HORIZON-EURATOM-2021-NRT-01-01, "Safety of operating nuclear power plants and research reactors".
- Key Objective of SASPAM-SA:
  - *Investigate the applicability and transfer of the operating large-LWR reactor knowledge and know-how to the near-term deployment of integral PWR (iPWR), in the view of Severe Accident (SA) and Emergency Planning Zone (EPZ) European licensing analyses needs.*
- Key Outcomes of SASPAM-SA :
  - *To be supportive for the iPWR licensing process by bringing up key elements of the safety demonstration needed;*
  - *To speed up the licensing and siting process of iPWRs in Europe.*

SASPAM-SA

Key Highlights and Consortium

- Key Highlights:
  - The applicability of large-LWR reactor knowledge & know-how to the near-term deployment iPWR, in view of SA and EPZ analyses, will be assessed and consolidated;
  - The research priorities will be identified in terms of methodology, code development and experimental needs;
  - The knowledge gained can support Regulators in decision-making as well as Industry and TSOs in assessing the applicability of iPWR safety features.
- The project is coordinated by ENEA and twenty-three Organizations from fourteen Countries are involved: ENEA, CIEMAT, CNRS, EDF, FZJ, GRS, INRNE, IRSN, KIT, KTH, LEI, POLIMI, RATEN, RUB, SINTEC, SSTC-NRS, SURO, TRACTEBEL, TUS, UNIROMA1, VTT, PSI, JRC.
- Project coordinator: Fulvio Mascari (fulvio.mascari@enea.it)

ENEA

GENERIC IPWR DESIGN CONSIDERED FOR THE ANALYSES

- To maximize the knowledge transferability and impacts of the project two generic design-concepts, characterized by different evolutionary innovations in comparison with larger operating reactor, have been selected for the analyses.

- These two generic reactor concepts include the main iPWR design features, considered in the most promising designs ready to go on the European market, allowing to assess in a wider way the capability of codes (SA and CFD) to simulate the SA phenomena typical of iPWR.

- It is not the project's objective to assess the generic reactor designs selected but, based on the project findings, allow a more general statement on the code's applicability to currently favored designs under postulated SA condition.

GENERIC IPWR DESIGN CONSIDERED FOR THE ANALYSES

- The activity here presented has been developed in the frame of the WP2, Input deck development and hypothetical SA scenarios assessment (SCENARIOS) coordinated by KIT, having as a major objective the development of generic but representative iPWR SA code input-decks and the analysis of the iPWR behavior under hypothetical postulated SA conditions.

- The scenarios identified are useful for the analyses of the severe accident code capability in modelling iPWRs and the evaluation of bottlenecks (if any)

- The activity is developed in collaboration with UNIROMA1 and CIEMAT;

- The studied plant is a generic iPWR characterized by:
  - The use of several passive systems;
  - Dry containment;
  - Electric power of about 300 Mwe;
  - In the framework of the SASPAM-SA project is named Design 2.

- The integral RPV contains, other than the core, all the major RCS components, which include:
  - Reactor coolant pumps (RCPs),
  - Compact Steam Generator (SG), e.g. helical coil bundles once-through SG,
  - Pressurizer (PRZ) and heaters.

GENERIC IPWR DESIGN CONSIDERED FOR THE ANALYSES

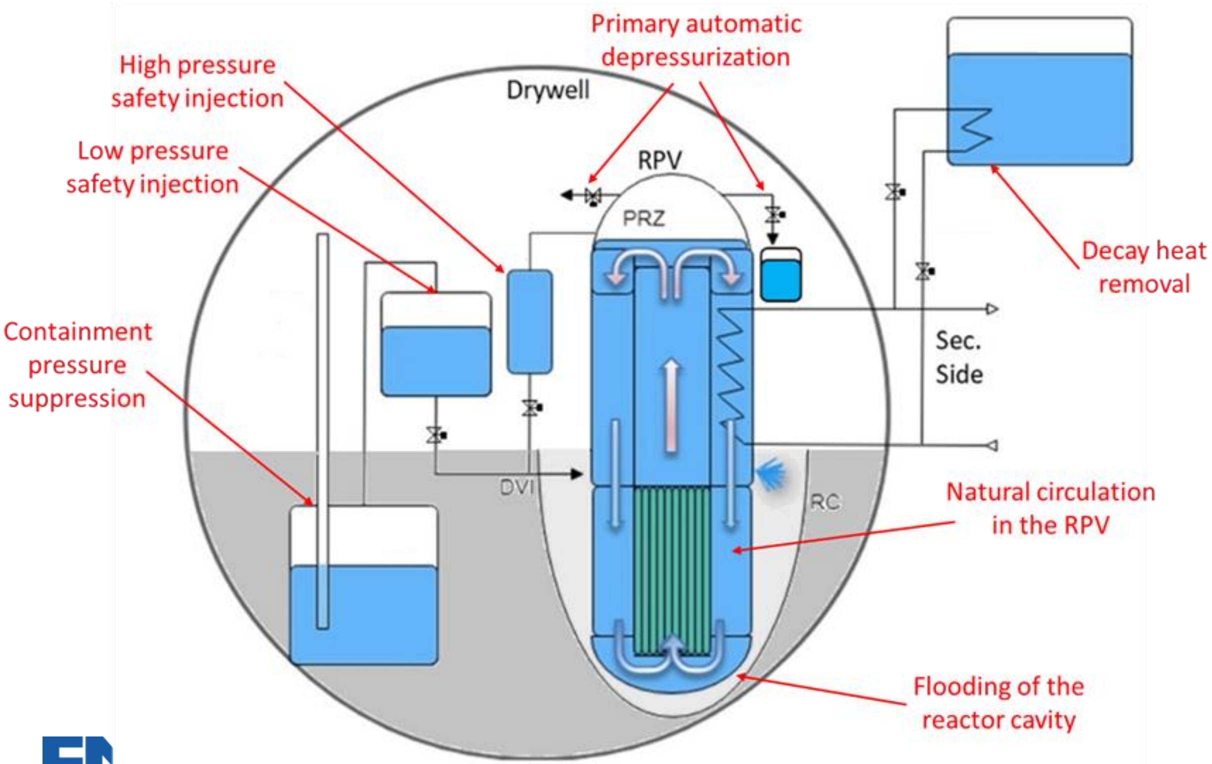
- Therefore, Design 2 is a generic reactor concept that could be representative of reactor concepts of interest.

- The Design 2 configuration permits the elimination of the occurrence of some accident sequences:
  - Large Loss Of Coolant Accident (LOCAs) are eliminated by the integral layout of the RPV
  - Only Small break LOCA can take place and will be studied.

- Since the aim of the activity is to test the capability of the MELCOR code in reproducing the behavior of different passive safety systems and their interaction in a SMR configuration, in order to analyze MELCOR capability in a wider way the following passive systems have been considered:
  - Decay heat removal;
  - Primary automatic depressurization;
  - High pressure safety injection;
  - Containment pressure suppression;
  - Low pressure safety injection;
  - Flooding of the Reactor Cavity (RC).

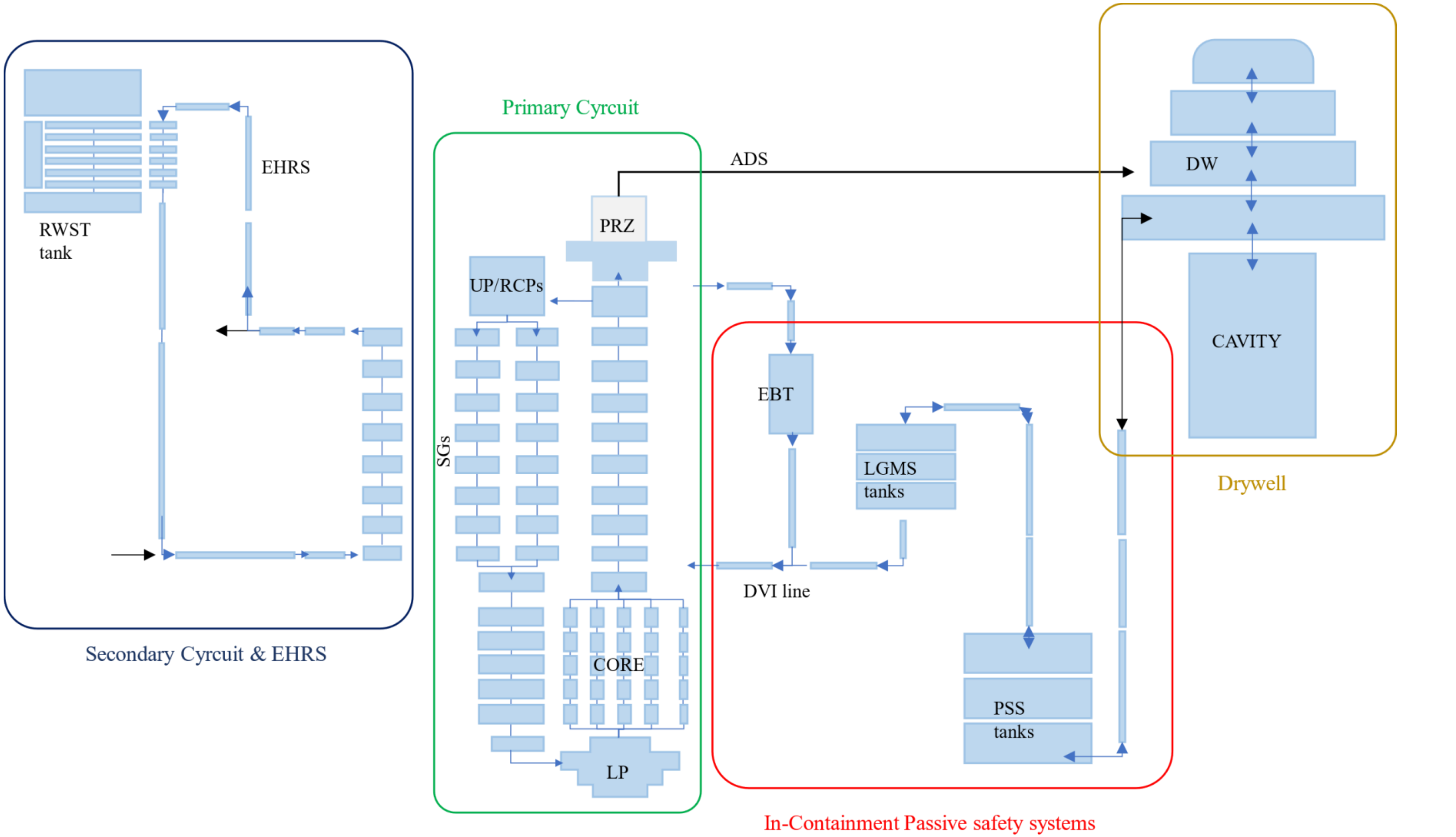
GENERIC iPWR DESIGN CONSIDERED FOR THE ANALYSES

- In order to develop the MELCOR input-deck, a reference database was needed.
- Considering the characteristics of Design 2 reactor type and the selected passive systems, a generic IRIS SMR type has been considered as reference for this analysis.
- During the MELCOR nodalization of the generic IRIS design, no proprietary data have been used

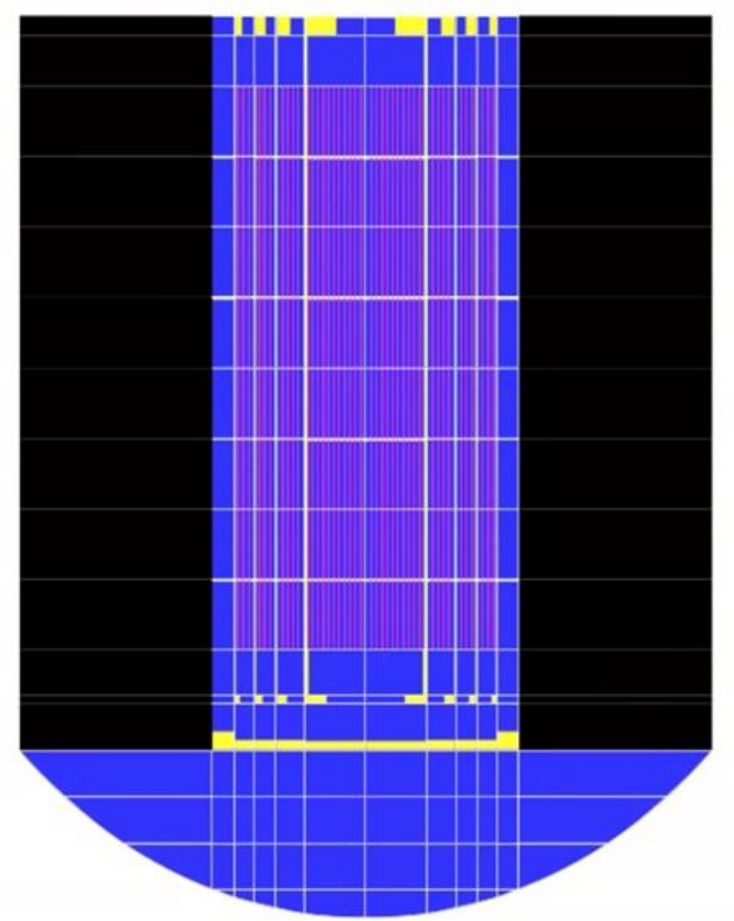


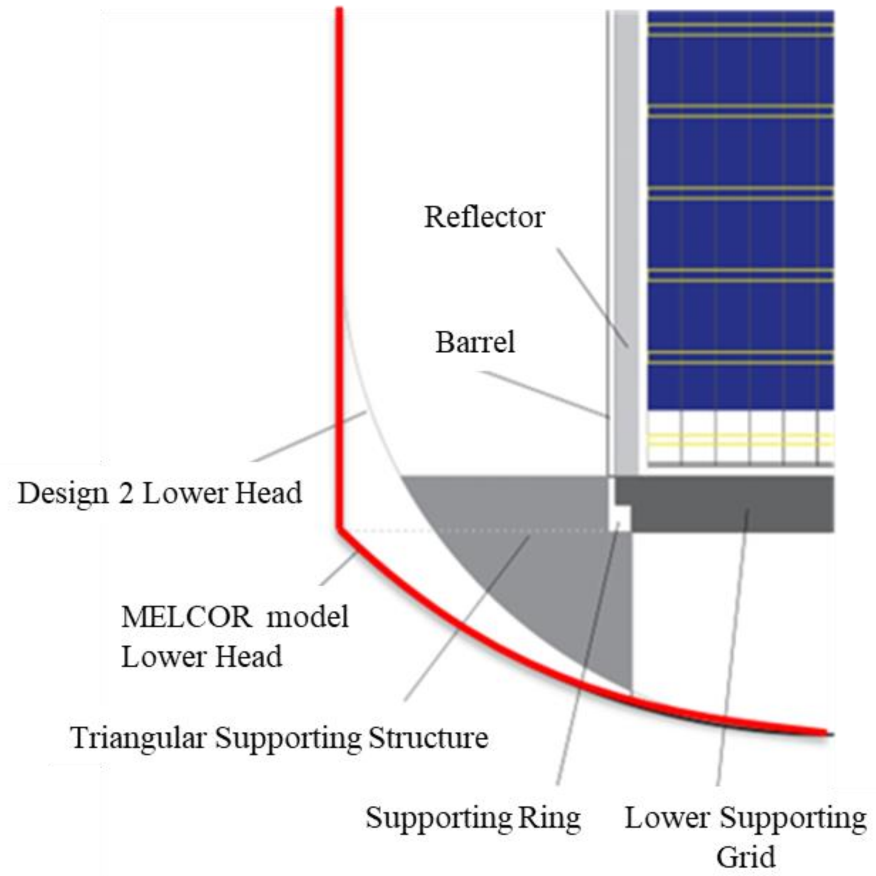
- Main geometric information has been determined by:
  - scaling the data available from the SPES-3 facility,
  - by engineering evaluation
  - public general data available for the IRIS reactor .

CVH NODALIZATION OF THE GENERIC IPWR CONSIDERED

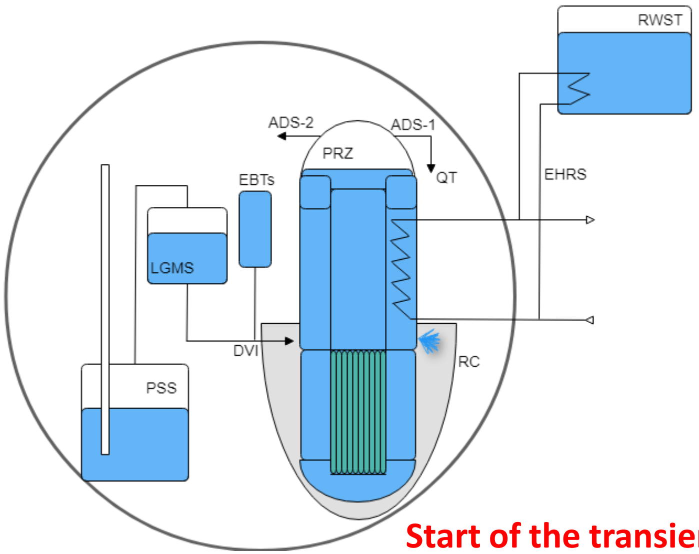


COR PACKAGE NODALIZATION OF THE GENERIC IPWR CONSIDERED

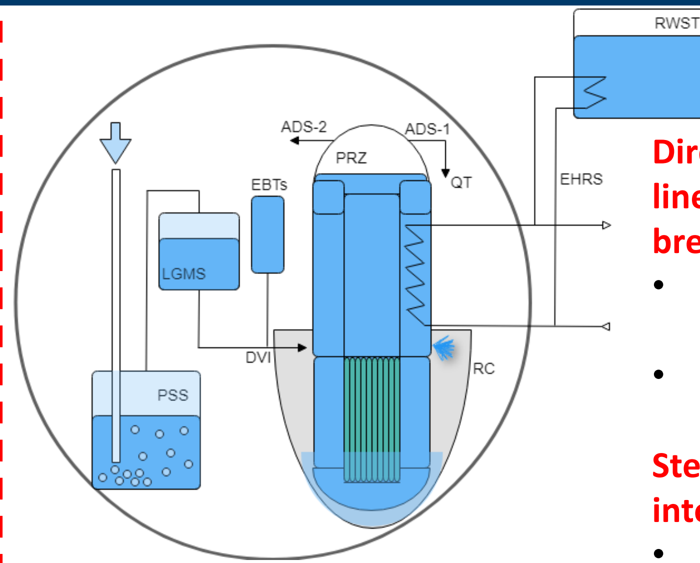




DBA PHENOMENOLOGY



Start of the transient (SOT) t=0

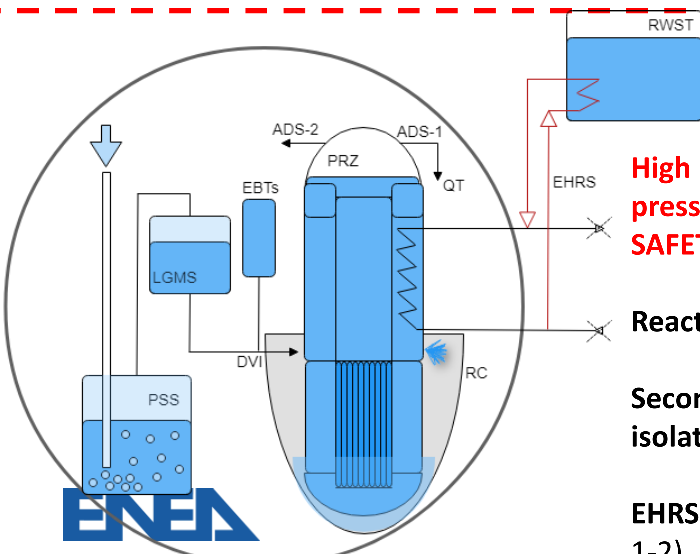


Direct Vessel Injection line (DVI) double ende break, d = 2 in

- RPV blowdown an depressurization;
- Containment pressurization.

Steam and gas dumpi into the PSS-vent pipe

- PSS, LGMS pressurization.

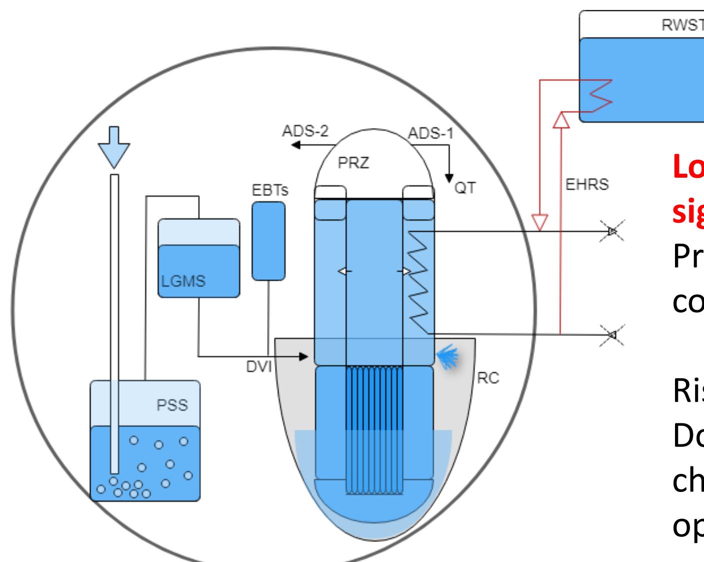


High containment pressure set point -> SAFETY signal

Reactor SCRAM

Secondary system isolation

EHRS activation (loop 1-2)

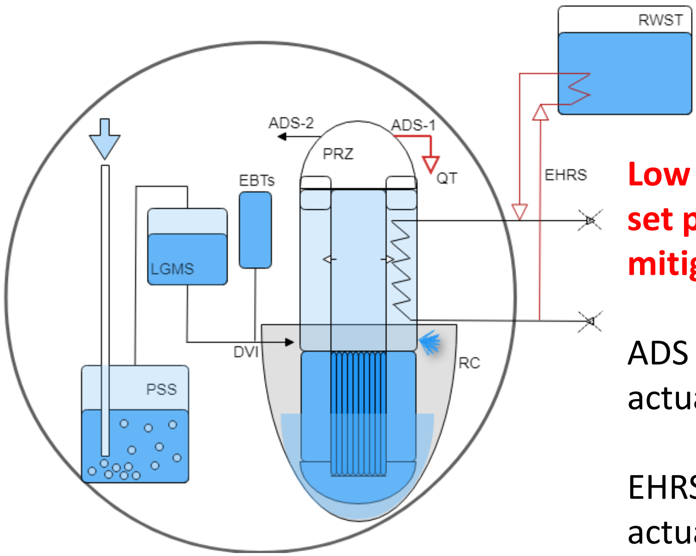


Low PRZ level signal:

Primary pumps coastdown

Riser-Downcomer check valves opening

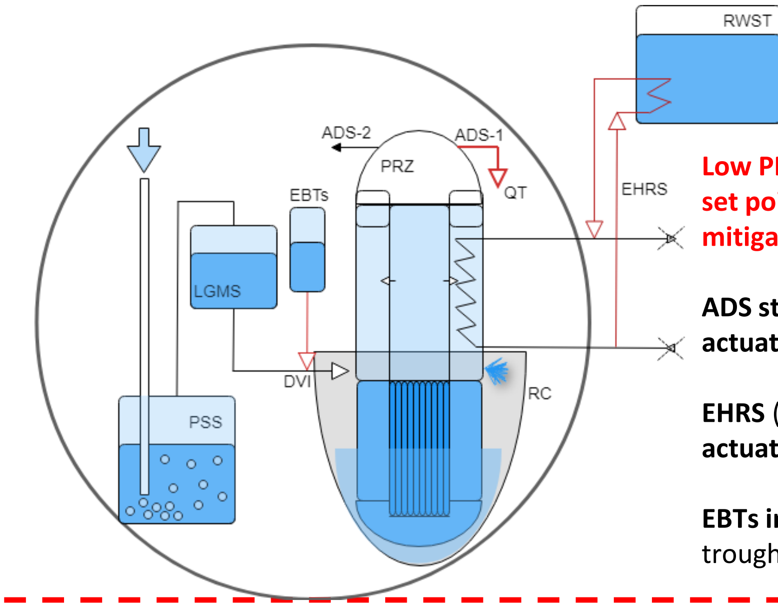
COR PACKAGE NODALIZATION OF THE GENERIC IPWR CONSIDERED



Low PRZ pressure set point -> LOCA-mitigation signal

ADS stage-1 actuation

EHRS (loop 3-4) actuation

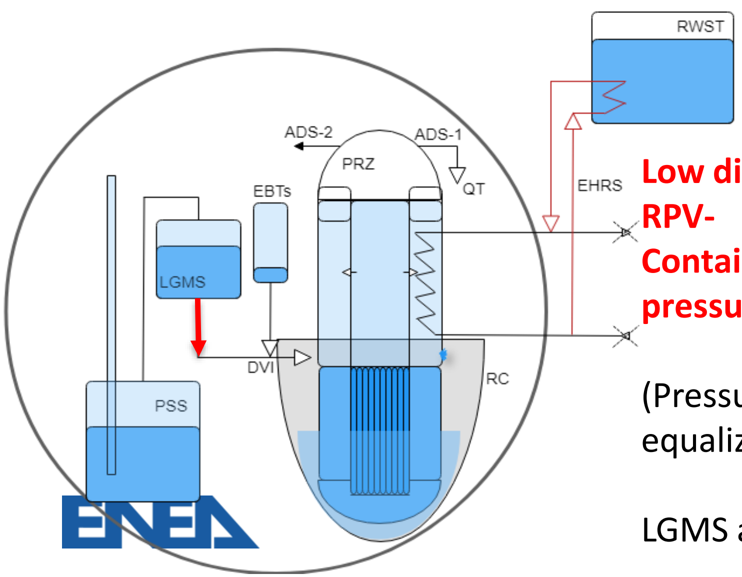


Low PRZ pressure set point -> LOCA-mitigation signal

ADS stage-1 actuation

EHRS (loop 3-4) **actuation**

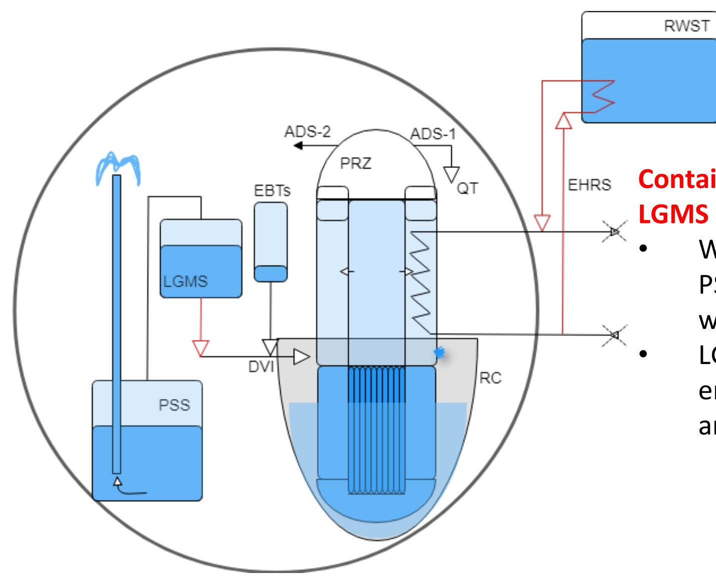
EBTs injection trough DVI intact line



Low differential RPV-Containment pressure signal

(Pressure equalization)

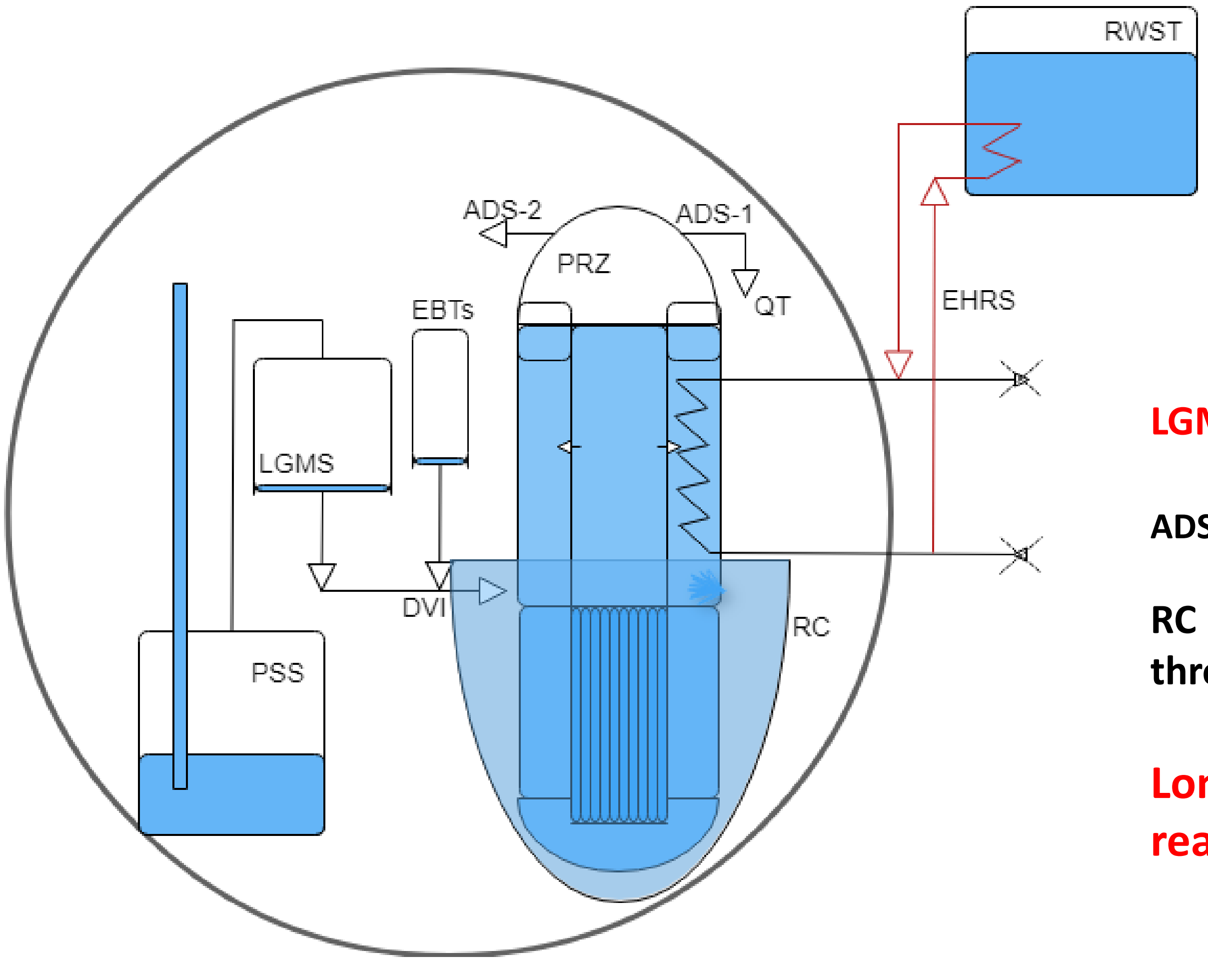
LGMS actuation



Containment, RPV – PSS, LGMS pressure inversion

- Water is pushed into PSS vent pipe -> water head pressure
- LGMS injections enhanced (into RC and RPV)

COR PACKAGE NODALIZATION OF THE GENERIC IPWR CONSIDERED



LGMS low mass signal

ADS stage-2 opening

RC is floaded -> Reverse flow through DVI lines into RPV

Long term cooling of the floaded reactor due to EHRS

PREDICTED PHENOMENA BY MELCOR CODE SIMULATION

System/Component	Predicted Phenomena by MELCOR simulation
RPV	Single and two-phase natural circulation; DVI break critical flow; Heat transfer in SG primary side; Heat transfer in covered and partially uncovered core
SG	Heat transfer in SG secondary side (boiling, dryout, superheating)
EHRS	Steam condensation in heat exchanger (heat transfer in tubes side); High and low pressure heat exchange in exchangers; Two-phase natural circulation
RWST	Natural convection in the pool; Thermal stratification during PhW1; Heat transfer pool side
EBT	Gravity driven injection
LGMS	Gravity and differential pressure driven injection
PSS	Influence of non-condensable gas on condensation heat transfer; Direct condensation for liquid-steam interaction; Thermal stratification
Drywell	Thermal stratification; Condensation on containment with presence of non-condensable; Heat losses to the environment; Thermal connection between RPV and Drywell
Reactor Cavity	Gravity driven injection
Quench Tank	Direct Condensation for Liquid-Steam interaction

DBA CALCULATION RESULTS

- SBLOCA analyzed, concerning the double-ended break of one DVI lines:
  - Lowest possible elevation for a LOCA and hence the most challenging accident scenario in terms of safety.

- Considering the DVI line-A as the broken line, the accident consists of 2-inch equivalent double-ended guillotine break of the DVI line.

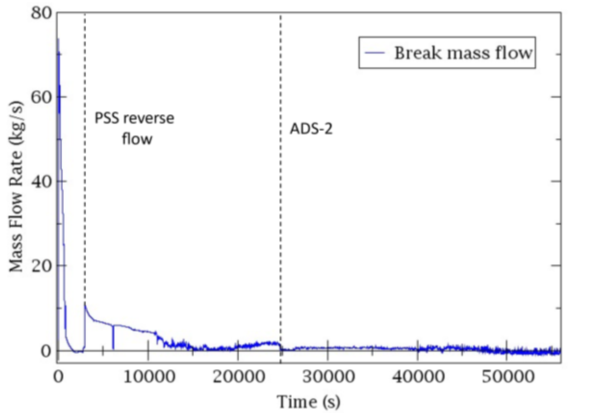
- Availability of all the emergency passive safety systems is assumed.

SEQUENCE OF MAIN TRANSIENT EVENTS

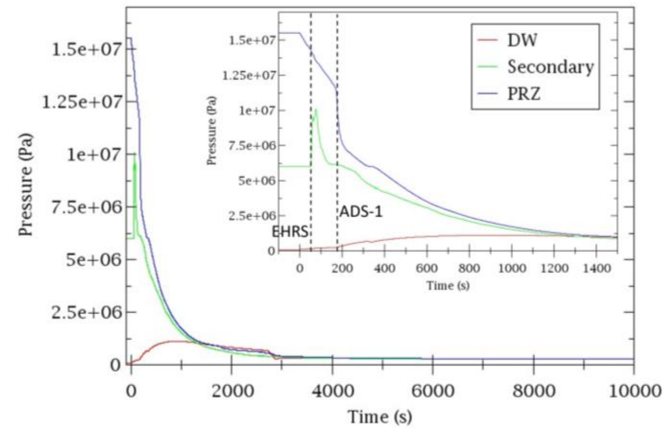
Main event	Signal	Time (s)	Actuation
Double guillotine break opening	\	0	
High containment pressure set point	HCP signal, S-Signal	52	SCRAM; Secondary syst. isolation; EHRS
Low pressurizer level set point	LPL signal	93	RCPs coast-down
	(HCP + LPL)	108	Shroud RI-DC valves open
Low pressurizer pressure set point	LPP signal	165	
	LM signal (HCP + LPP)	170	EBT line opening
		173	ADS stage-1 opening
EBT-A emptying	\	1120	
Low differential pressure RV-Containment	LDPC signal	1386	LGMS starts to inject
Reverse flow PSS-DW	\	2712	
EBT-B emptying	\	3960	
Low LGMS mass set-point	LCC signal	24689	ADS stage 2
LGMS-A emptying	\	34240	
Flow from RC to RPV	\	37480	
LGMS-B emptying	\	55090	

DBA CALCULATION RESULTS

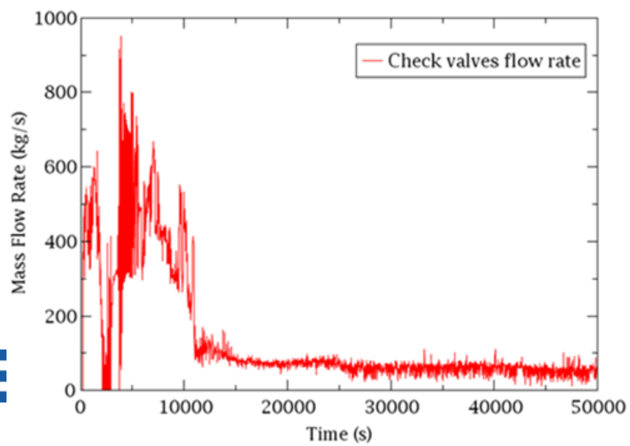
Mass flow rate through break



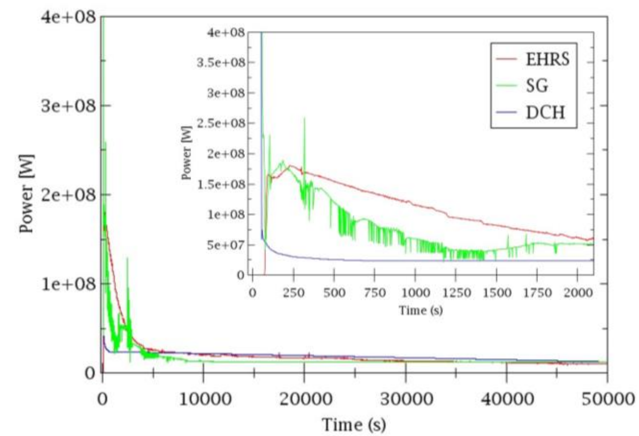
Pressure evolution in PRZ secondary system and DW



Mass flow rate though the RI-DC check valves

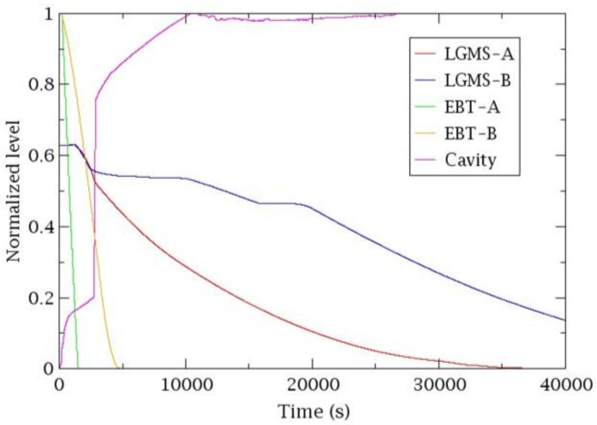


Decay heat, SGs power from RPV to EHRS loops, power from EHRS to RWST

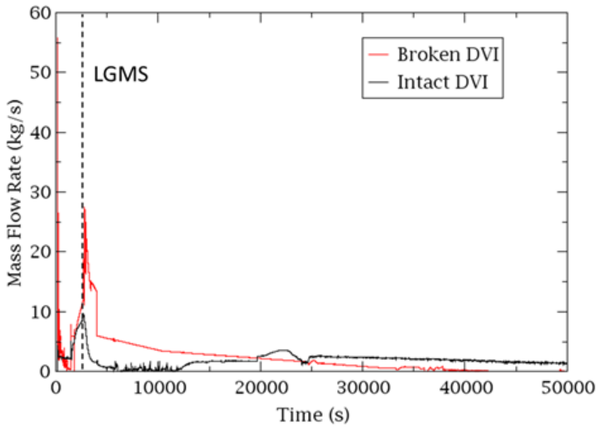


DBA CALCULATION RESULTS

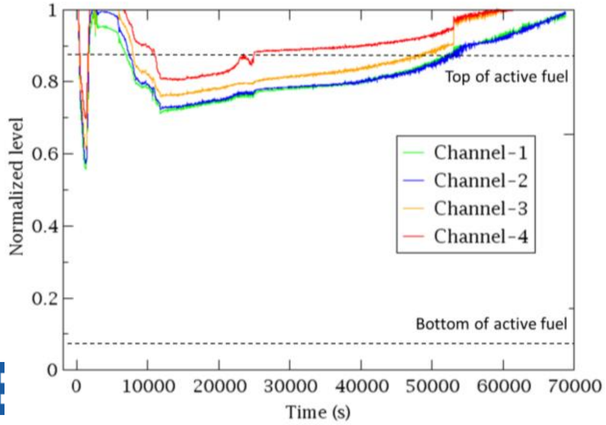
Normalized water level inside the tanks of LGMS (A and B), EBT (A and B) and reactor cavity.



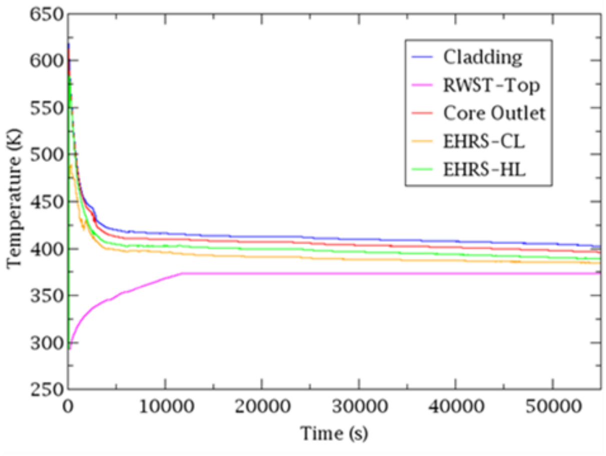
Mass flow rate though the broken DVI (to the RC) and the intact DVI (to RPV).



Normalized levels inside the core channels (between upper and lower nozzle grids).



Temperature of cladding, EHRS hot leg, outlet of core and RWST



FIRST CONCLUSIONS

- The present SMR accident analysis, developed in the SASPAM-SA Horizon Euratom project, provided interesting information about the MELCOR integral code capability to simulate the associated DBA transient phenomena in a SMR using a passive mitigation strategy, as starting point for future postulated plausible BDBA and severe accident sequences.

- The overall accident evolution is in accordance with the expected plant behavior and the phenomenological windows are reproduced.

- The MELCOR model was able to reproduce the reference steady state conditions.

- Analyzing the simulation results of the DBA transient it can be concluded that the code can simulate:
  - Thermal-hydraulic coupling between containment and the RPV,
  - Passive safety systems.

  this permits the passive mitigation strategy of the reactor, leading to the RPV depressurization and core cooling in safety conditions.

FIRST CONCLUSIONS

- The specific phenomena that take place in the different passive systems are predicted, e.g.:
  - Steam condensation in heat exchanger,
  - Natural convection in the pool;
  - Thermal startification in passive system

- The phenomenology predicted by MELCOR are consistent with other studied presented in public literature.

- In order to assess quantitatively the result a validation activity against selected phenomena is necessary.

- Further derivative activities focused on assessing the capability of MELCOR code to simulate the phenomena taking place in postulated plausible severe accident scenarios in iPWR are planned.

HORIZON EURATOM SASPAM-SA PROJECT

Acknowledgement/Disclaimer

SAFETY ANALYSIS OF SMR WITH PASSIVE MITIGATION STRATEGIES - SEVERE ACCIDENT

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ENEA ACTIVITIES IN THE FRAMEWORK OF IAEA CRP I31033

ENEA ACTIVITIES IN THE FRAMEWORK OF IAEA CRP I31033

- In the framework of the IAEA CRP “Advancing the State-of-Practice in Uncertainty and Sensitivity methodologies for Severe Accident Analysis in Water-cooled reactors”, a MELCOR input-deck of the QUENCH-06 has been developed.

- Goals of the activity are:

  - Indipendent user validation of Core Heat up, Zircaloy-Steam Oxidation and Degradation models embedded into best estimate MELCOR code employing the experimental dataset provided by QUENCH-06 test;
  - Sensitivity analysis (SA) adopting several Zircaloy-Steam oxidation reaction rates;
  - Uncertainty analysis (UA).

- A dedicated presentation has been done during the EMUG22.

The activity has been done in collaboration with Politecnico di Milano.

ENEA ACTIVITIES IN THE FRAMEWORK OF IAEA CRP I31033

TEST Control Volumes (CV)

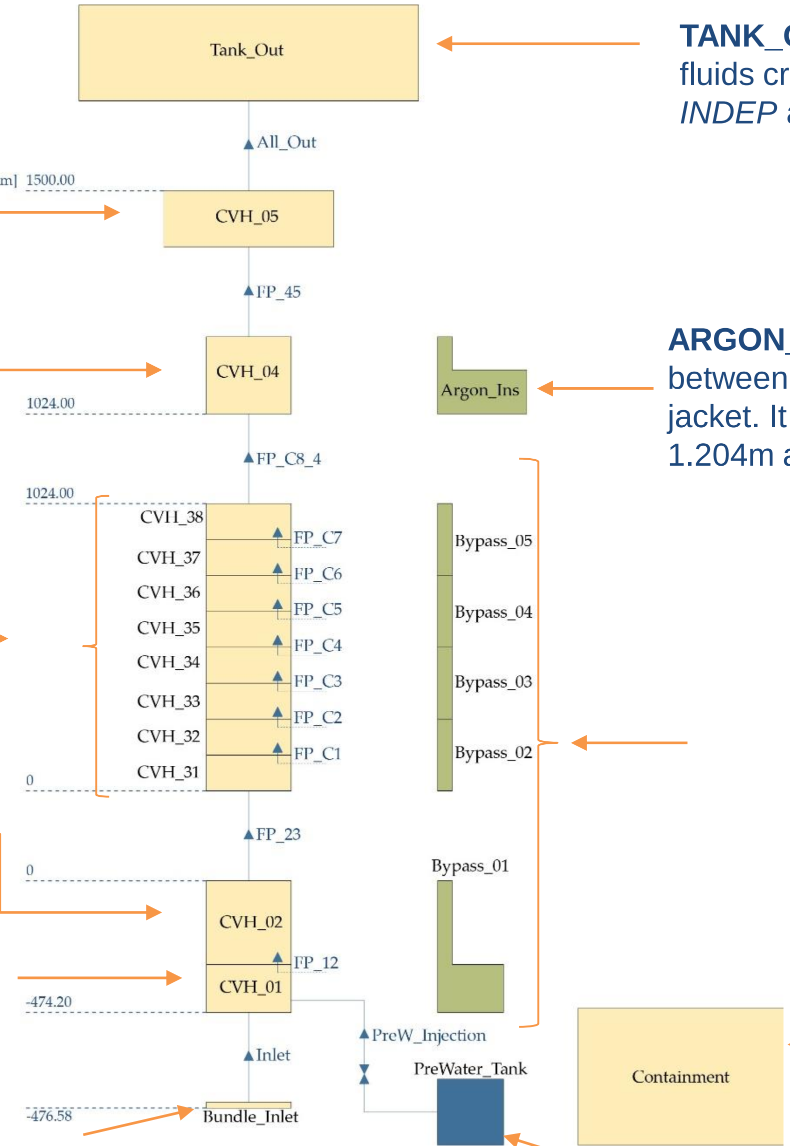
CVH_05: : CV in corrispondence of upper Mo electrode.
- Post FWI Argon mass flow rate source

CVH_04: : CV in corrispondence of upper Mo electrode

CVH_31-38: : CVs in active region

CVH_02: : CV in correspondence of lower Mo electrode

CVH_01: CV in correspondence of Cu electrode
- Steam, Argon and quenching water flow rates sources



TANK_OUT: it acts as a sink for all fluids crossing the test section (*TIME – INDEP* at 2 bar, 600 K)

ARGON_INS: argon insulation between the shroud and the cooling jacket. It acts also as bypass between 1.204m and 1.3m.

CONTAINMENT: Argon filled *TIME-INDEP* volume at 415 K.

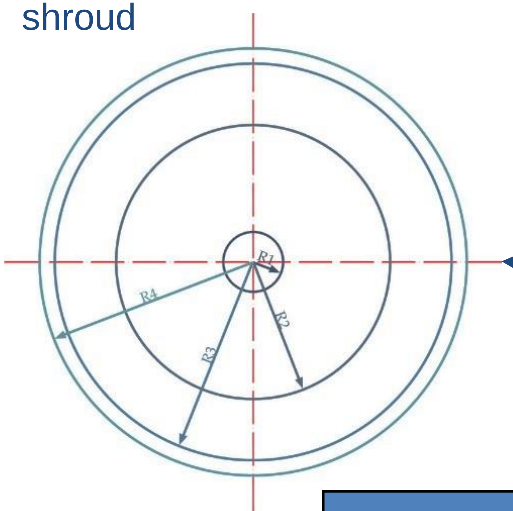
PREW_TANK: FWI system

ENEA ACTIVITIES IN THE FRAMEWORK OF IAEA CRP I31033

Test Bundle up to zircaloy shroud is nodalized in *4 concentric rings and 42 axial levels*.

RADIAL NODALIZATION

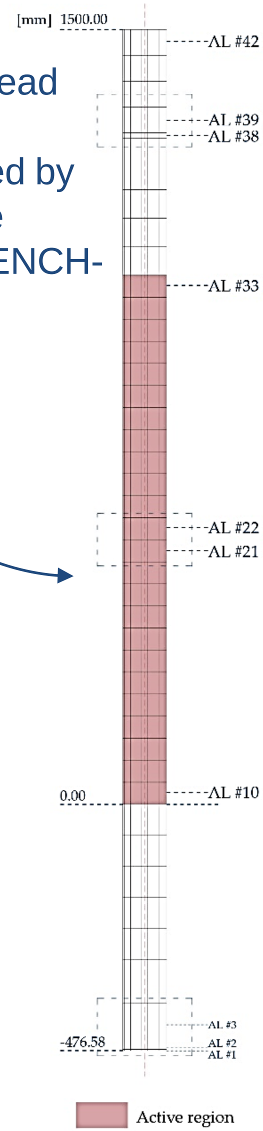
- *Ring 1*: unheated rod + grids
- *Ring 2*: 8 inner heated rods + grids
- *Ring 3*: 12 outer heated rods + grids
- *Ring 4*: 4 corner rods + grids + shroud

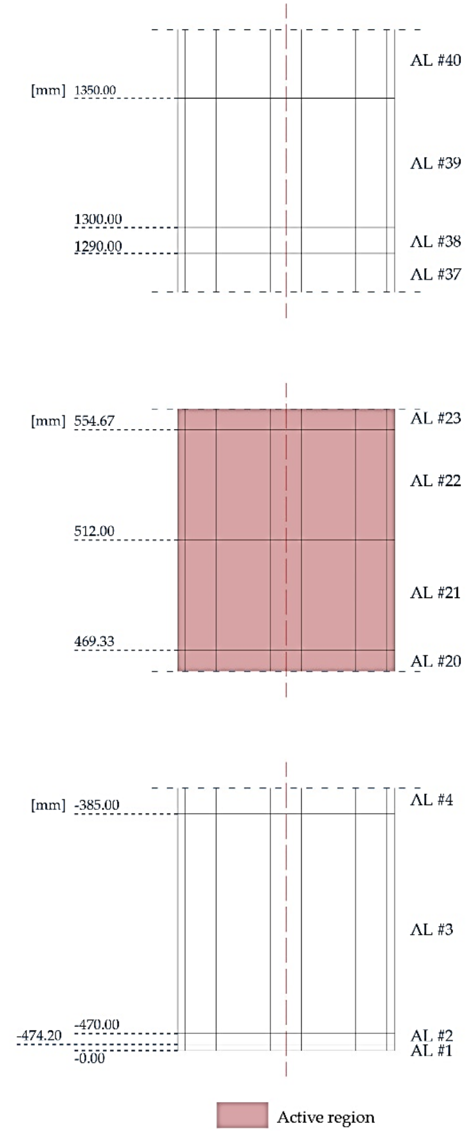


Ring	Radius
Ring #1	6 mm
Ring #2	27.975 mm
Ring #3	39 mm
Ring #4	42.38 mm

AXIAL NODALIZATION

- *Axial level 1*: lower head (dummy)
- *Axial level 2*: occupied by core supporting plate
- *Axial level 3-42*: QUENCH-06 test bundle





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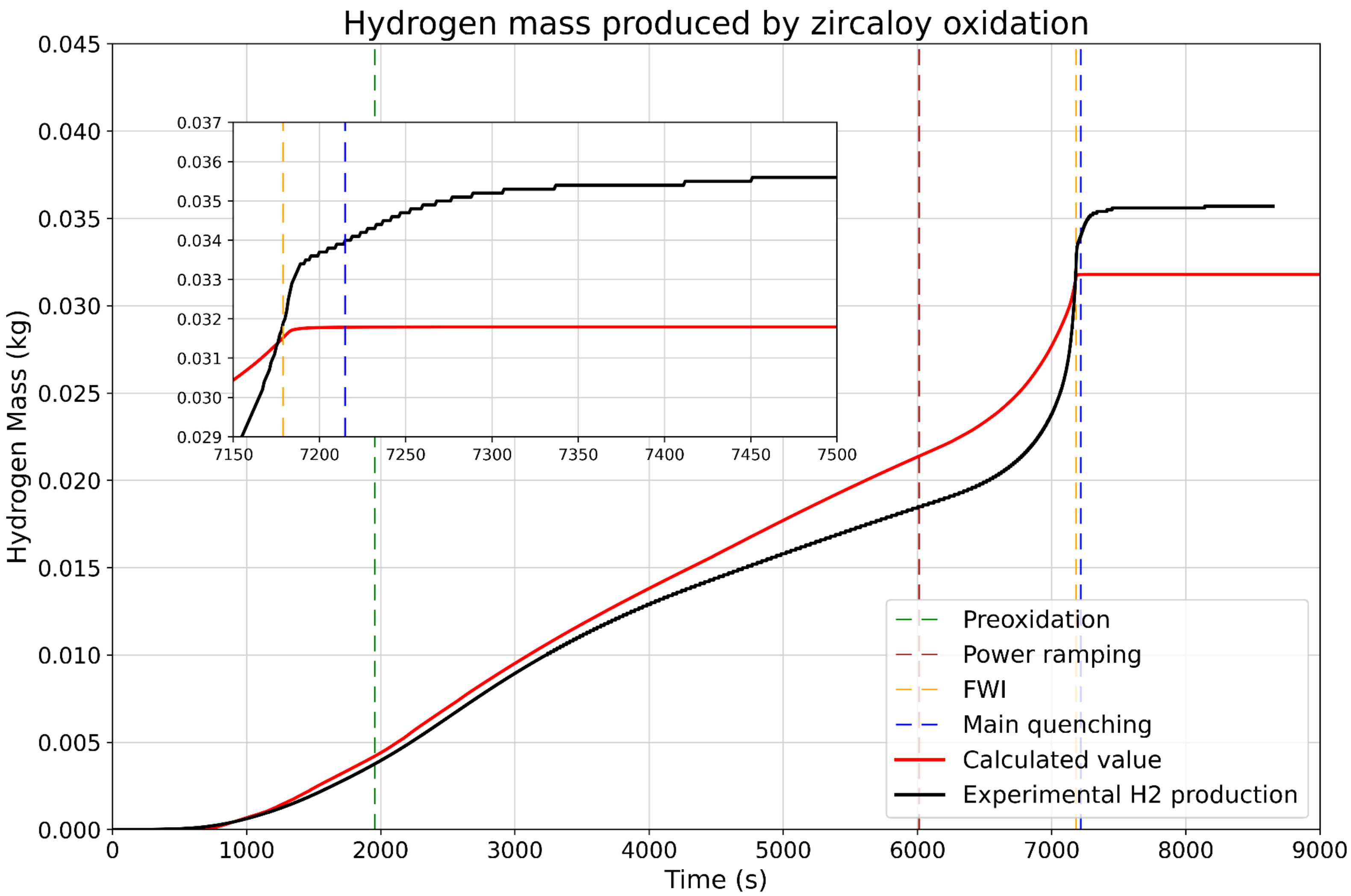

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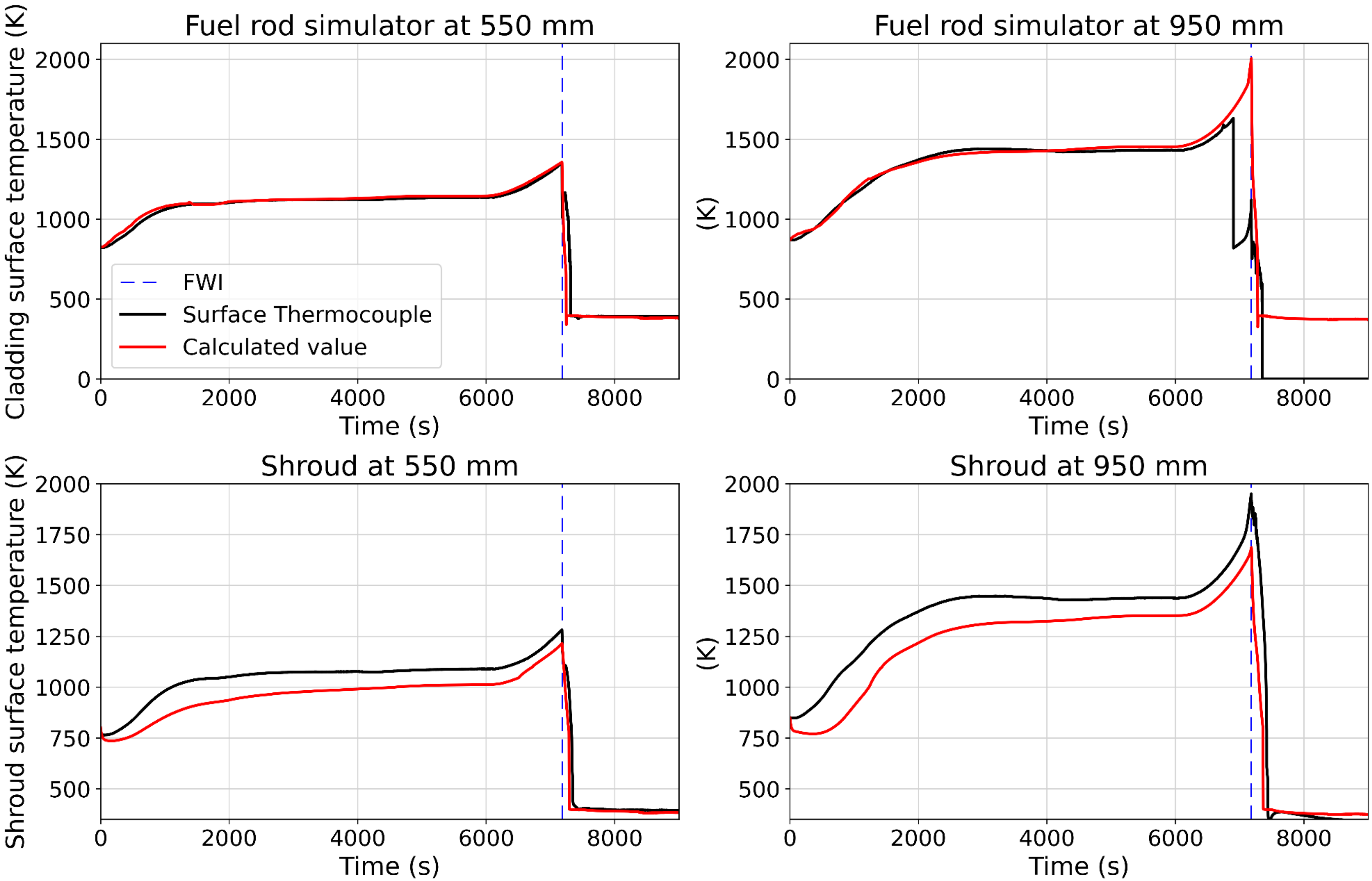

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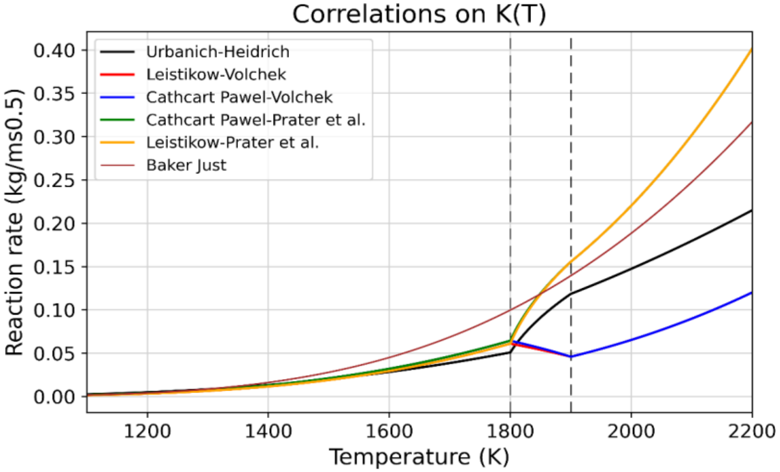
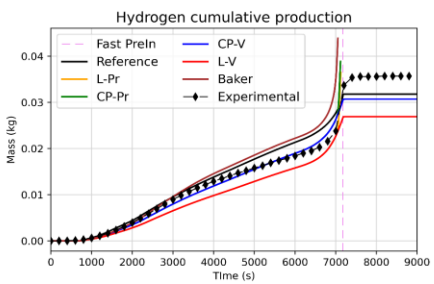
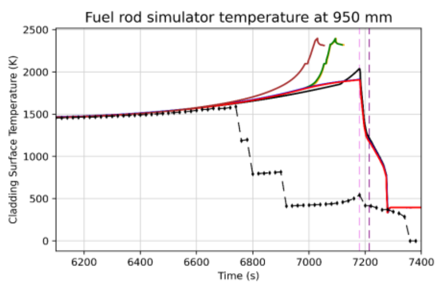
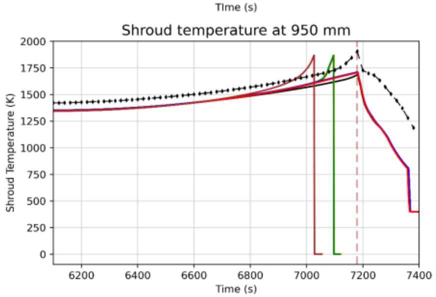
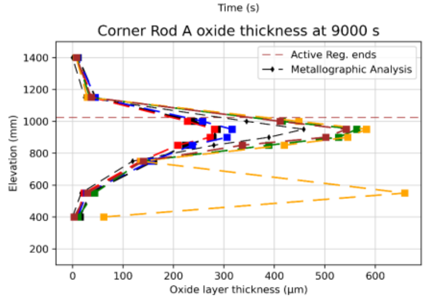
SENSITIVITY ANALYSIS



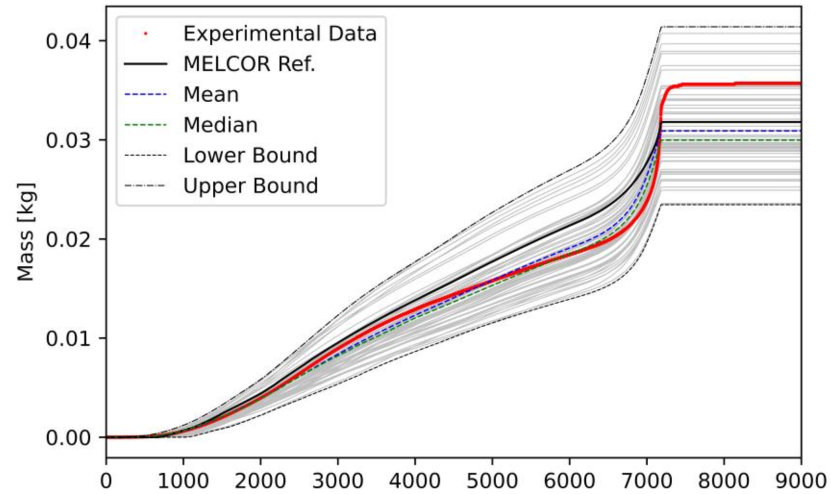


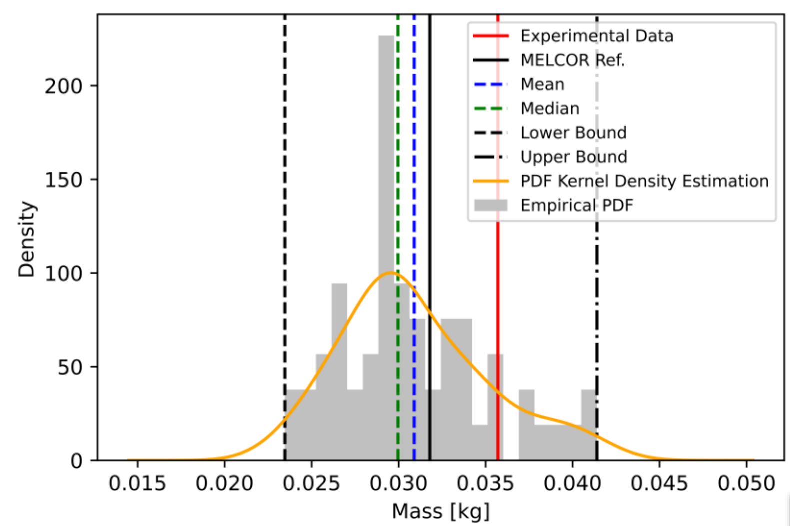






UNCERTAINTY ANALYSIS





THANK YOU FOR YOUR KIND ATTENTION

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