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

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

11th Meeting of the European MELCOR and MACCS User Group, Paul Scherrer Institute, Villigen, Switzerland, April 3-5, 2019

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ENEA ACTIVITIES IN THE FIELD OF SEVERE ACCIDENTS

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

ENEA ACTIVITIES IN THE FIELD OF SEVERE ACCIDENTS

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

- 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.

ENEA ACTIVITIES IN THE FIELD OF SEVERE ACCIDENTS (UNCERTAINTY)

- 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 next research activity that could developed in domestic and in international framework *(e.g. MUSA project funded in the H2020 European Framework Programme, etc.…),* ENEA is starting different activities related to the uncertainty estimation:
  - is developing uncertainty analyses using the DAKOTA software tool coupled with MELCOR code in SNAP environment/architecture;
  - In collaboration with the Sapienza University of Roma is developing uncertainty analyses using the RAVEN software tool coupled with MELCOR code;
  - In collaboration with the Politecnico di Torino is developing uncertainty analyses using the DAKOTA software tool coupled with TRACE code in SNAP environment/architecture (TH activity).
  - Is in progress to start an analogous activity in collaboration with IRSN using the SUNSET software tool coupled with ASTEC code.

ENEA ACTIVITIES IN THE FIELD OF SEVERE ACCIDENTS (FUSION REACTOR)

- 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 distribuetd.

ENEA ACTIVITIES IN THE FIELD OF SEVERE ACCIDENTS (FUSION REACTOR)

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			

ENEA ACTIVITIES IN THE FIELD OF SEVERE ACCIDENTS

- EU project JASMIN (Joint Advanced Severe Accidents Modelling and Integration for Sodium-Cooled Fast Neutron Reactors):
  - ENEA uses 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 project CESAM (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 submitted for the publication.

GRAZIE PER LA VOSTRA ATTENZIONE

ENEA PLANT CALCULATIONS ACTIVITY IN THE FRAMEWORK OF THE EU-IVMR PROJECT

ENEA ACTIVITY IN THE IVMR PROJECT

IVMR Project of H2020 European Program "In-Vessel Melt Retention Severe Accident Management Strategy for Existing and Future NPPs"

- The goal of the project is an analysis of the applicability and technical feasibility of the IVMR strategy to high power reactors, both for existing ones as well as for future reactors of different types (PWR, BWR and VVER).

- The main outcomes of the project will be: relevant assumptions and scenarios to estimate the maximum heat load on the vessel wall, additional experimental data, improved numerical tools for the analysis of IVMR issues and a harmonized methodology on the IVMR.

ENEA ACTIVITY IN THE IVMR PROJECT- REACTOR CALCULATION

- FIRST STEP → Most critical scenarios have been identified for each reactor type and the possibility of IVMR has been assessed, using current models and codes.
  - ENEA has performed IVMR calculations for the **PWR900-like** reactor using ASTEC V2.1.0.3 code (LBLOCA, SBLOCA and SBO sequences) and **MELCOR 2.1 (LBLOCA sequence).**
- SECOND STEP → Improved models and updated codes will be used to re-assess the possibility of success of IVMR.
  - The presented ASTEC calculations [[study carried out with ASTEC V2, IRSN all rights reserved, (2019)],] have been performed with:
    - ✓ the V2.1.1.1 release of the code and
    - ✓ with an improved input data deck with respect to the 1st set of calculations carried out, at the beginning of IVMR, with a previous ASTEC release (V2.1.0.3). Main input data deck improvements:
      - Activation of the model dealing with the heat transfer by radiation from the corium in the lower plenum;
      - Adoption of a more refined radial meshing of the vessel wall.
    - ✓ New ASTEC calculations are in progress with the last ASTEC release (V2.1.1.4) and they will be completed before the end of IVMR project (final set of calculations) .
  - ENEA, after a review of the MELCOR nodalization performed in the EU-CESAM and EU-FASTNET project, is performing the updated analyses with last releases of MELCOR 2.2 (2019 updated results).
  - The analysis of the <u>reference case is in progress and sensitivity calculations are in progress to analyses the effect of selected parameters on the LP corium behavior</u>.

ENEA IVMR CODE APPLICATIONS

- The SBO accident scenario has been also simulated with MELCOR code using, as well as possible, the same conditions of ASTEC calculation.

- LB-LOCA, SB-LOCA and SBO accident scenarios have been recalculated with the most recent release of ASTEC code (V2.1.1.1).

SBO CODE APPLICATION: SBO SCENARIO MAIN CHARACTERISTICS

- *SBO with failure of emergency diesels*

- SBO transient is unmitigated and it is characterized by:

  - ✓ Primary circuit depressurization at severe accident signal (Tcore out > 650°C);
  - ✓ Failure of reactor coolant pump seal injection;
  - ✓ Isolation of accumulators when P < 15 bar;
  - ✓ Loss of containment sprinkler;
  - ✓ Start of external vessel cooling 3 h (10800 s) after the accident start (before corium slumping into the lower-head.

- In MELCOR code, for a first analyses, the cavity has been filled with water from the beginning of the transient.

CODE NODALIZATIONS: MELCOR MODEL OF THE PWR900-LIKE REACTOR

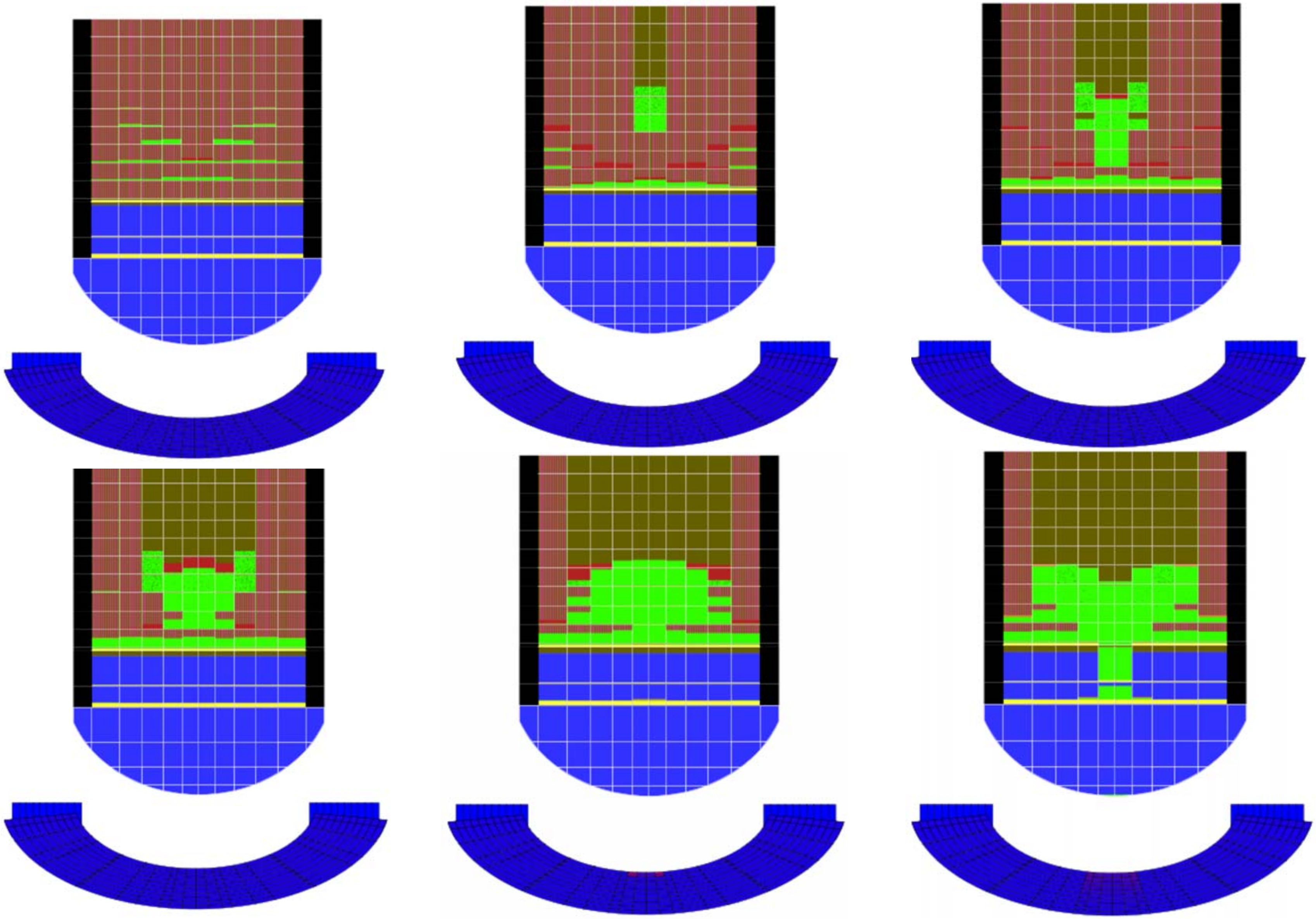
SOME MELCOR CODE OPTIONS USED BY THE USER:

➢ Heat transfer coefficient from in-vessel falling debris to pool:	2000.0 W/m2K
➢ Debris to Lower Head Heat Transfer:	100.0 W/m2K
➢ Velocity of falling debris:	0.01 m/s
➢ Porosity of particulate debris :	0.4
➢ Particulate debris equivalent diameter:	
• Core :	0.01 m
• LP :	0.002 m
➢ SC1250: COR package temp. for enhanced debris to lower head conduction:	2800 K

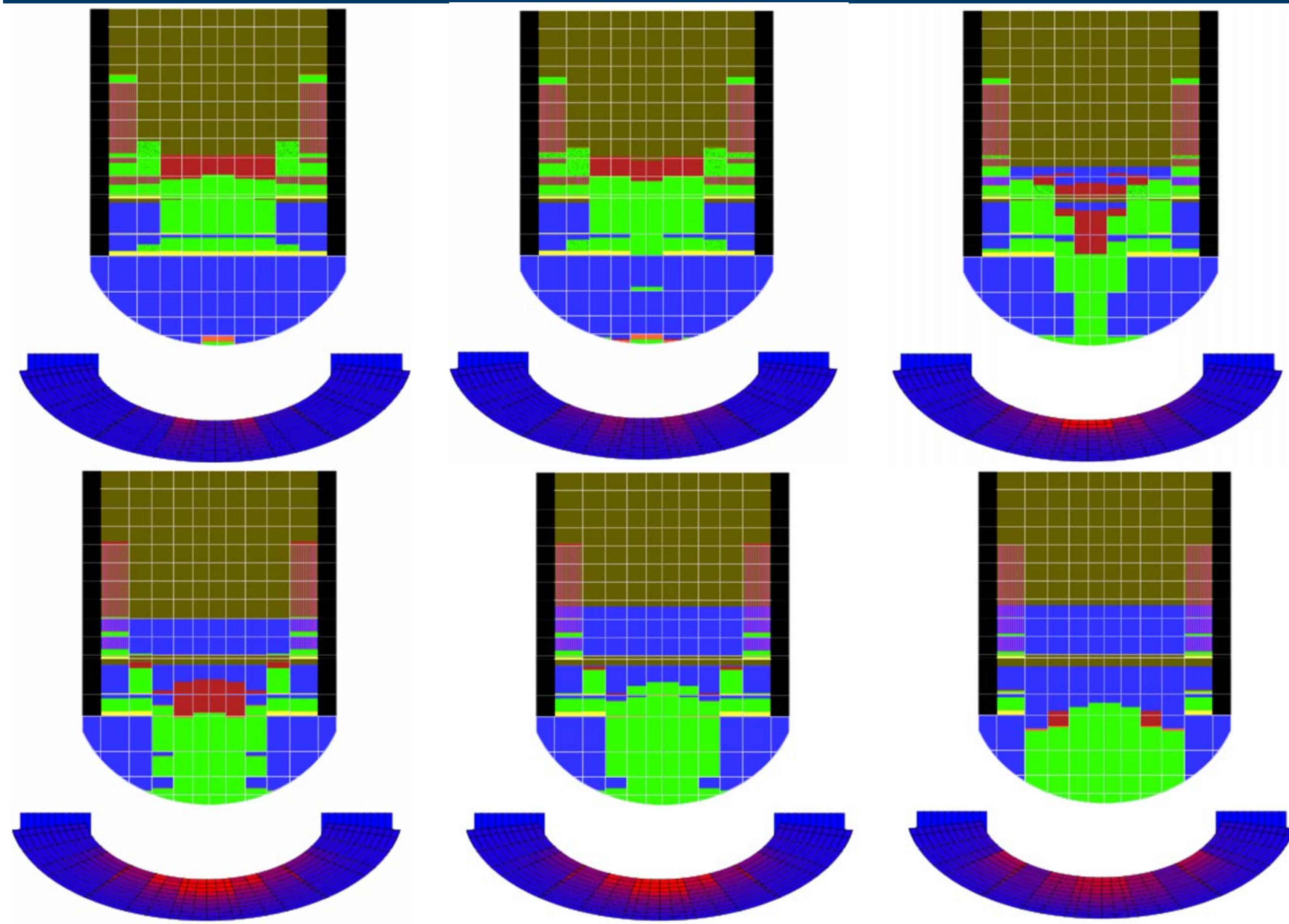
➢ Lower Head Failure Modeling Parameters :

- Option for heat transfer coefficient from the oxidic molten pool to the lower head: *The coefficient is to be calculated from the internal model*
- Option for heat transfer coefficient from the metallic molten pool to the lower head: *The coefficient is to be calculated from the internal model.*

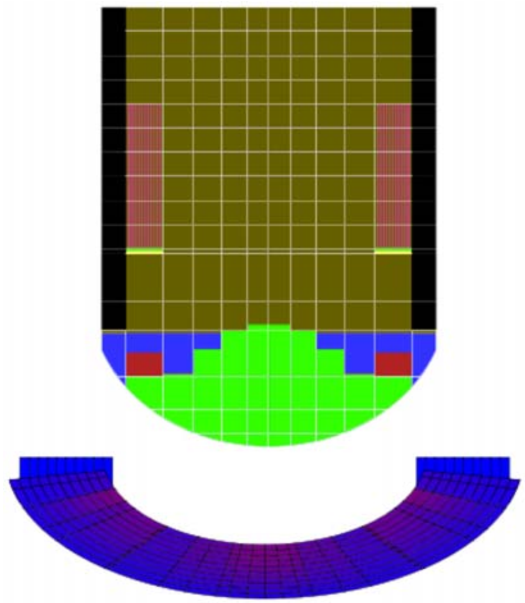
SBO CODE APPLICATION: MELCOR CORE DEGRADATION VISUALIZATION, BY USING SNAP, SELECTED INSTANTS

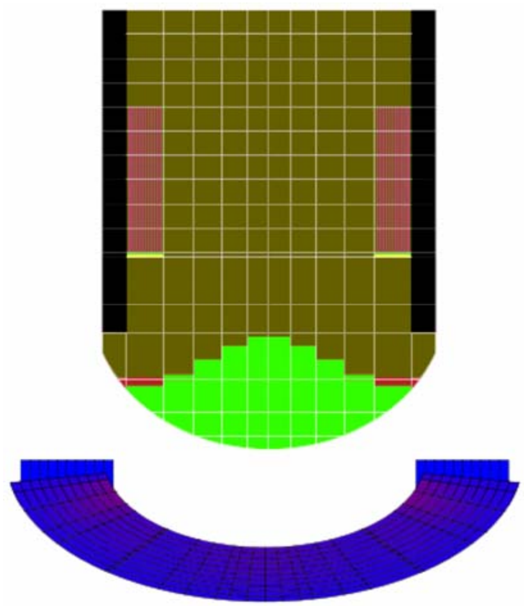


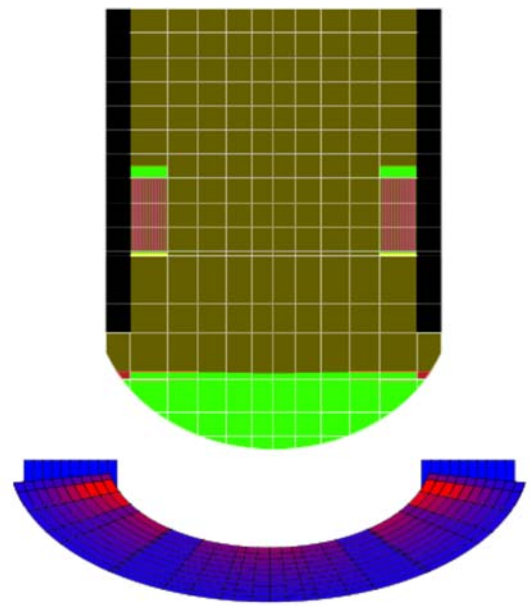
SBO CODE APPLICATION: MELCOR CORE DEGRADATION VISUALIZATION, BY USING SNAP, SELECTED INSTANTS (2)

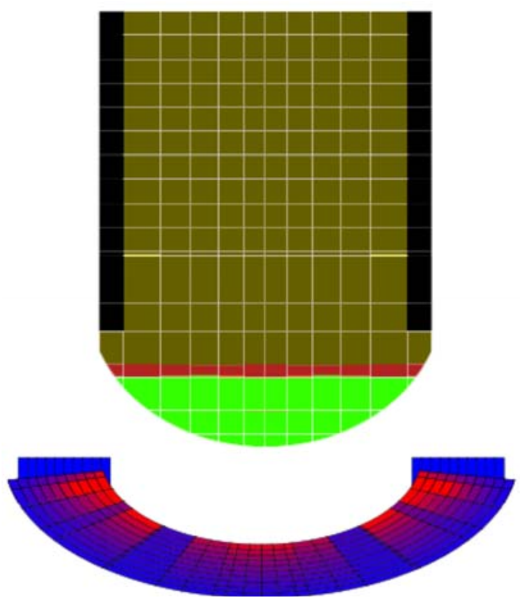


SBO CODE APPLICATION: MELCOR CORE DEGRADATION VISUALIZATION, BY USING SNAP, SELECTED INSTANTS (2)





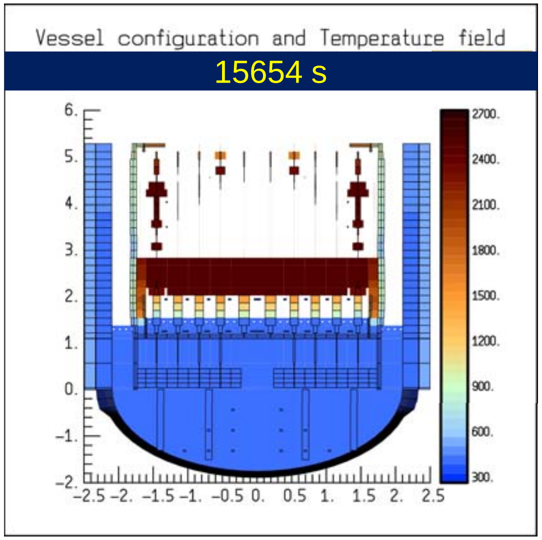


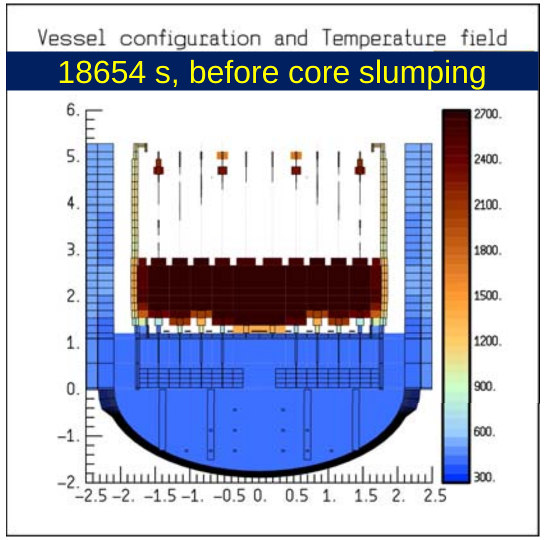


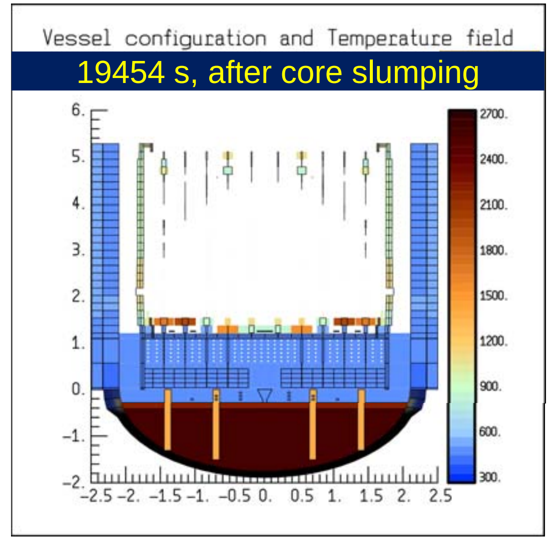
- In the last 3000 s of the transient the MELCOR calculation are characterized by bottom layer with particulate debris and top layer with metallic molten pool (small and intermittent quantity of oxidic molten pool at the top of the debris bed layer is observed)].

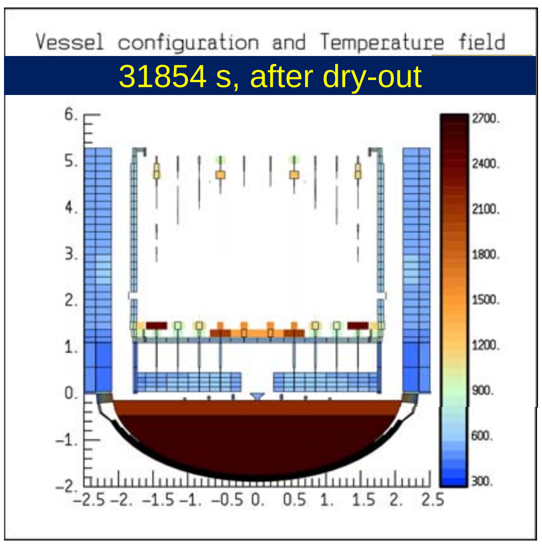
- Debris bed mass is important during the whole transient before and after slumping (UO2 and ZrO2 are dominant).

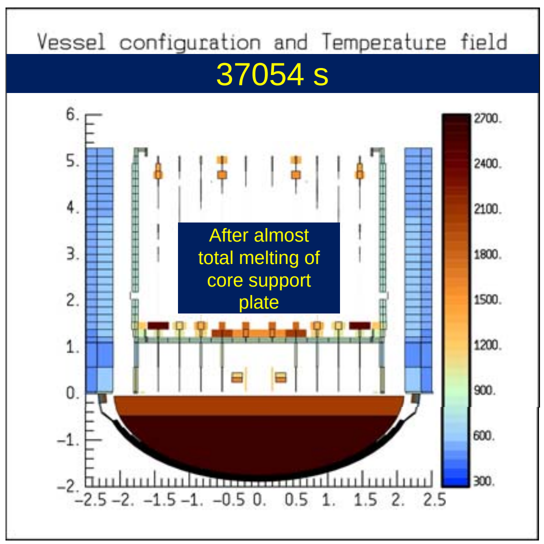
SBO CODE APPLICATION: ASTEC V2.1.1.1 CORE DEGRADATION VISUALIZATION, SELECTED INSTANTS







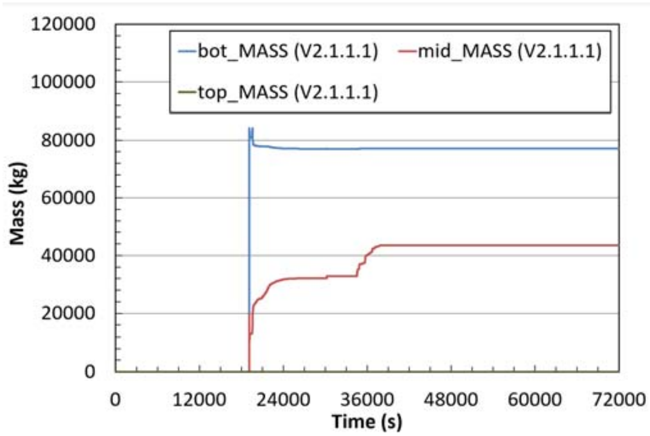




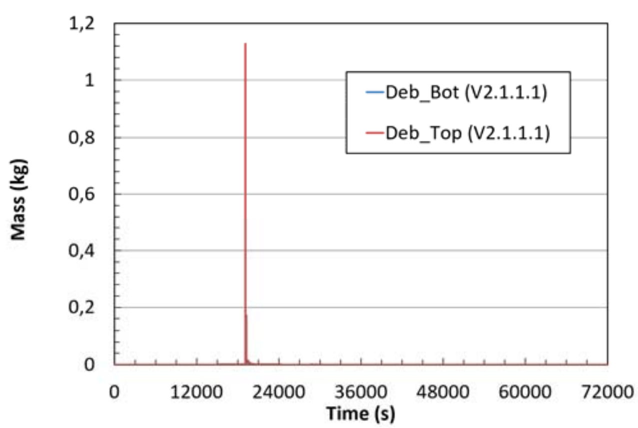
Corium pool composed of oxides at the bottom and metals at the top. Progressive increase of the top metals layer height due to melting of lower-head structures, ablation of vessel wall and almost total melting of core support plate (radiation from the top of molten corium).

SBO CODE APPLICATION: ASTEC VS MELCOR: CORIUM MASSES AND COMPOSITION IN THE LP

Corium pool mass in the 3 layers (ASTEC)

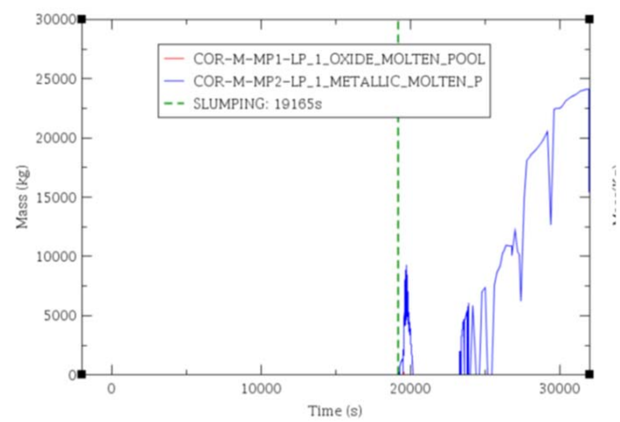


Debris bed mass at the top and bottom of corium pool (ASTEC)

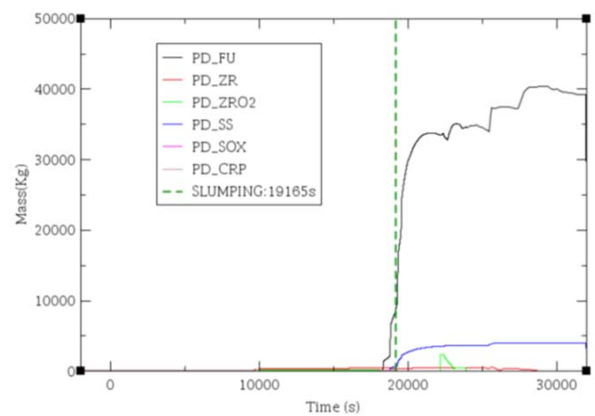


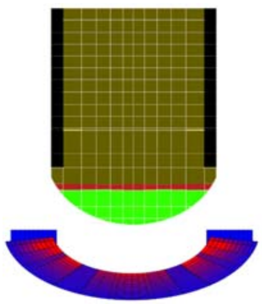
After slumping, the corium pool is composed of 2 layers during the whole transient in both ASTEC calculations (no mass in the 3rd top layer). Layers composition is presented in the next slide. Debris mass is negligible or absent.

Oxide and metallic molten pool mass (MELCOR)



Debris bed bottom mass (MELCOR)

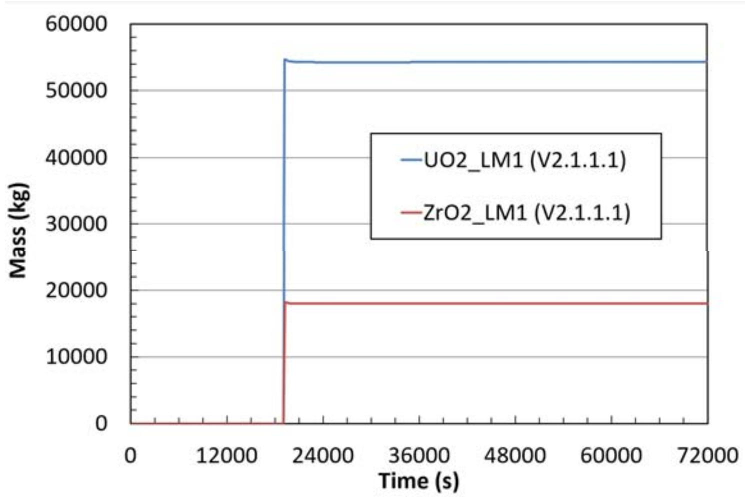




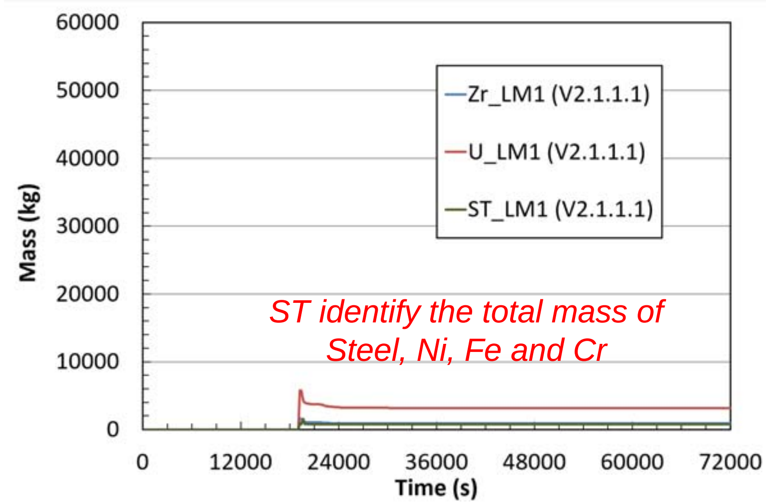
In MELCOR, LP till vessel failure is filled at the top with metallic molten pool with a small and intermittent quantity of oxidic molten pool. Debris bed mass is important during the whole transient after slumping (UO_2 and ZrO_2 are dominant).

SBO CODE APPLICATION: COMPOSITION OF CORIUM POOL IN THE LP

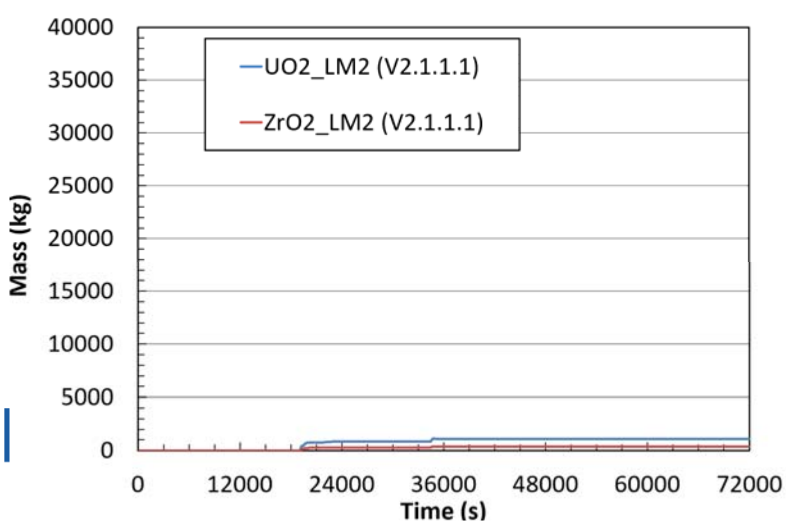
Oxides in the bottom corium pool layer



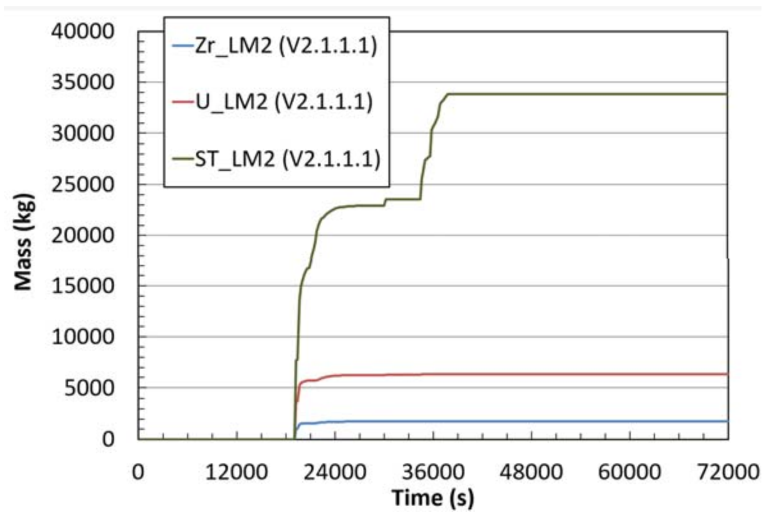
Metals in the bottom corium pool layer



Oxides in the middle (top) corium pool layer

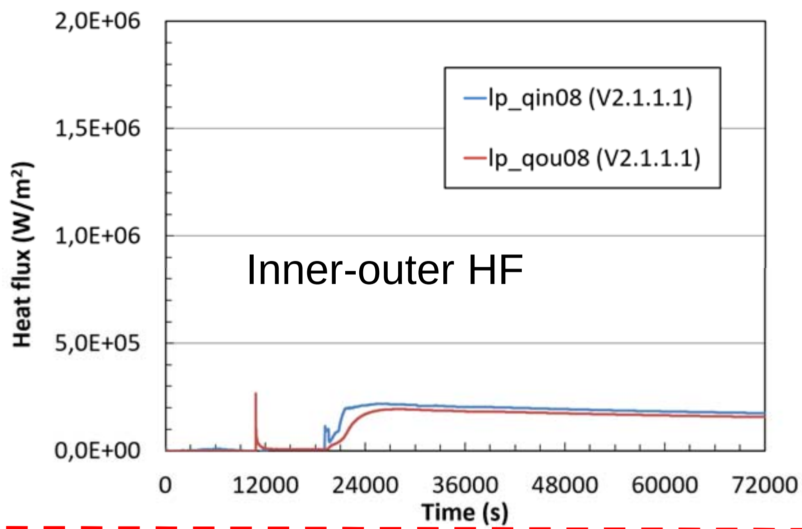


Metals in the middle (top) corium pool layer

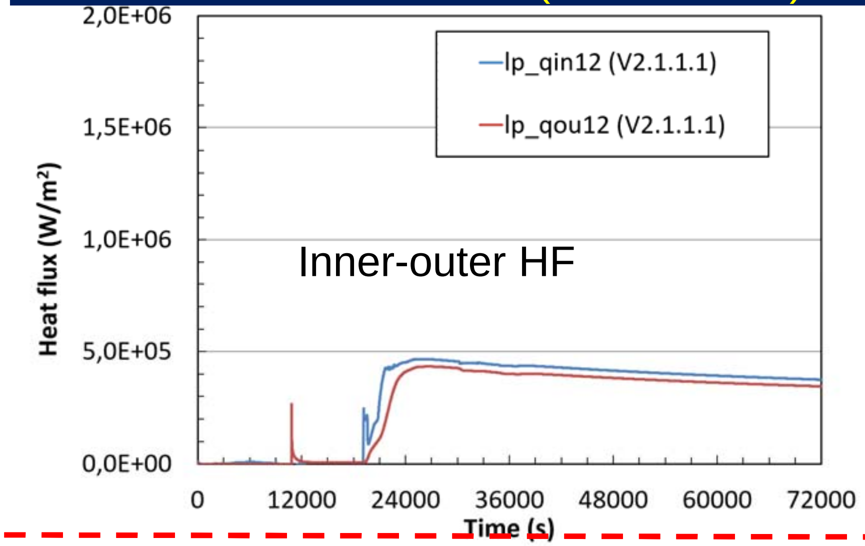


SBO CODE APPLICATION: ASTEC VS MELCOR LH HEAT FLUX VS. TIME

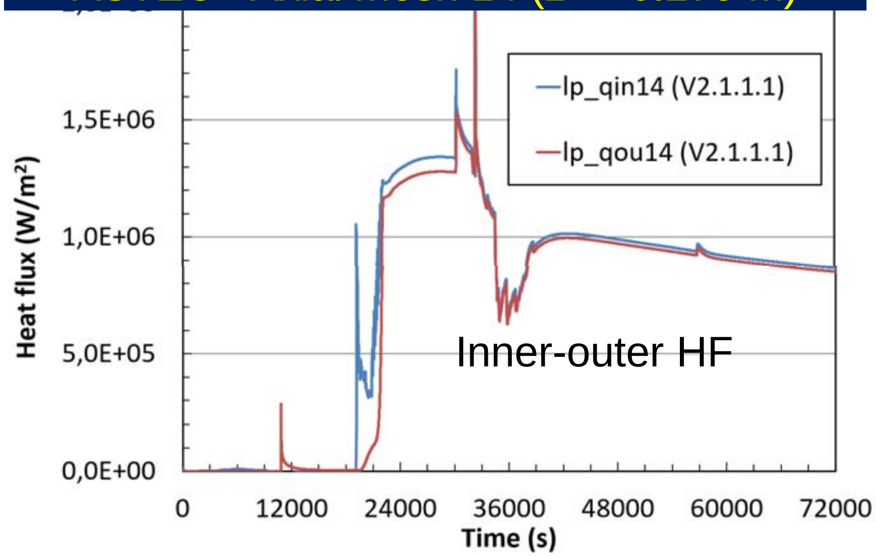
ASTEC - Axial mesh 8 (z = -1.242 m)



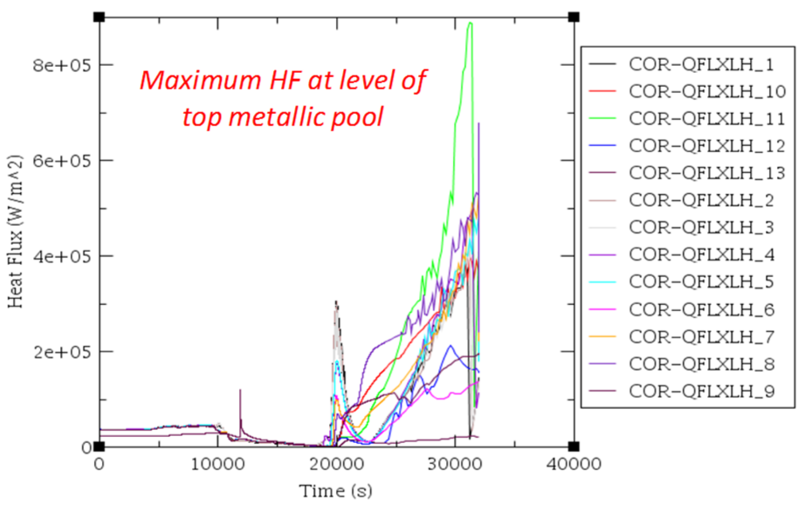
ASTEC - Axial mesh 12 (z = -0.629 m)



ASTEC - Axial mesh 14 (z = -0.275 m)



MELCOR - from the bottom to the top of LP



SBO CODE APPLICATION: ASTEC VS MELCOR LP HEAT FLUX VS. TIME

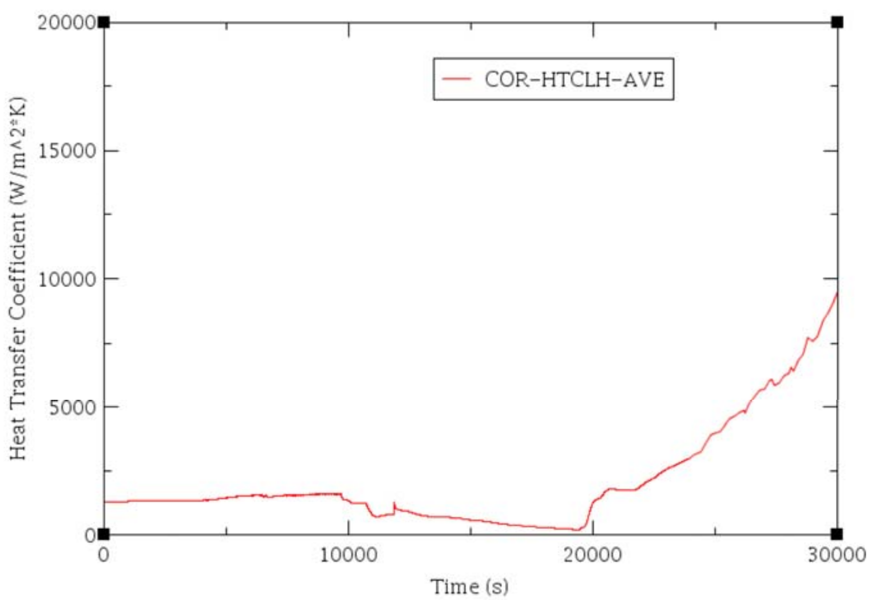
- Heat flux peaks can be observed in all calculations during very fast transients (as at the time of corium slumping in the LP).

- After the peak correlated with corium slumping the heat flux in MELCOR calculation increases at all LH axial levels until the vessel failure.

- The maximum heat flux (the peaks correlated with corium slumping or observed during very fast transients are not considered) is:
  - 0.89 MW/m2 (at roughly 31194 s, just before the vessel failure) in the MELCOR reference calculation.
  - 1.41 MW/m2 (at roughly 31000 s) in the ASTEC V2.1.1.1 calculation.

SBO CODE APPLICATION: ASTEC VS MELCOR SIMULATION OF EXTERNAL LH COOLING

HTC vs time in MELCOR



- In ASTEC calculations, the external vessel cooling is simulated with a boundary condition on the outer surface of the lower-head. The imposed HTC is 10000 W/m^{2}K (representative of nucleate boiling regime) and the external temperature is 110°C. The cooling starts after 3 h of accidental transient.

- In MELCOR calculation, the external cooling is simulated with cold water filling the cavity from the beginning of the transient.
    - ✓ The HTC is calculated by the code (see figure) and it is similar to the HTC imposed in the ASTEC calculation only at the end of calculated MELCOR transient.

CONCLUSIONS

- Calculation are in progress with the MELCOR 2.2 version;
- A reference calculation has been performed and the detail analyses is in progress;
- By a first comparison between ASTEC and MELCOR code results:
  - The sequence of main events is similar between ASTEC and MELCOR code;
  - The corium slumping into the LP take place at very similar times;
  - The time of the maximum heat flux in the LP take place at very similar times;
  - The maximum heat Flux predicted by MELCOR is at *of the top of metallic pool as ASTEC;*
  - The maximum heat flux predicted by MELCOR is smaller in comparison with ASTEC (0.89 MW/m^2 VS 1.41 MW/m^2)
  - The corium physical characteristics in the LP are significantly different:
    - 2 layers molten pool in ASTEC (oxides at the bottom and metals at the top) with negligible particulate debris;
    - Bottom layer with particulate debris and top layer with metallic molten pool in MELCOR [small and intermittent quantity of oxidic molten pool].

CONCLUSIONS

- Some sensitivity analyses are in progress to study the behavior of the corium in the lower plenum and the effect of some parameters in the MELCOR calculations
- The next calculation will be runned by assuming:
  - *Heat transfer coefficient from in-vessel falling debris to pool: 10 W/m2K*
  - *Debris to Lower Head Heat Transfer: 10 W/m2K*

- A paper, related to the ASTEC calculations, has been presented at the ERMSAR-2019 conference:

  - S. Ederli, F. Mascari, ASTEC simulation of In Vessel Retention Strategy applied to a generic PWR 900 Mwe, The 9TH European Review Meeting on Severe Accident Research (ERMSAR2019), Prague, Czech Republic, March 18-20, 2019.

- A paper, related to the MELCOR calculations and some comparison with ASTEC, is going to be submitted at the next NURETH-18.

Authors thank the funding received from the H2020 Framework Programme of the European Commission via the IVMR project;

Special thanks to IRSN for the coordination of Project;

Special thanks to JRC for the coordination of the WP2.5 activities

ITALIAN SAFETY ASSESSMENT ACTIVITY

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 .

- A complete review of the input deck has been performed and the In-vessel analysis with the code MELCOR 2.1 of three unmitigated LBLOCA severe accidents in a generic PWR of 900 MWe, caused by three distinct initiator events
  - Double-ended rupture of the cold leg of Loop 1;
  - Double-ended rupture of the hot leg of Loop 1;
  - Double-ended rupture of the surge line;

  The activity has been conducted in the ENEA in the framework of a Master Degree thesis with the University of Bologna.

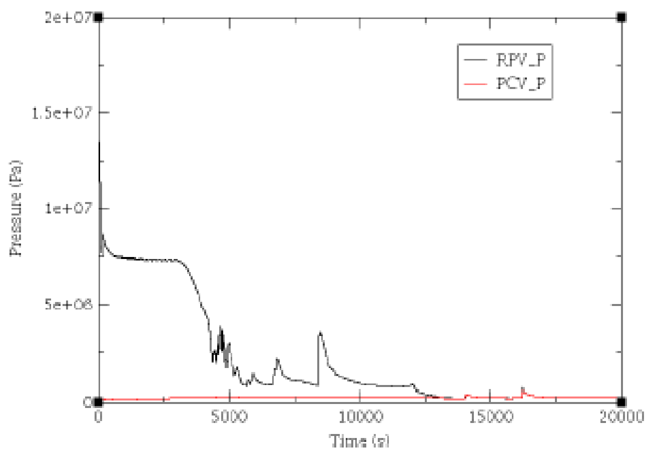
- *The results of the activity have been presented at the:*

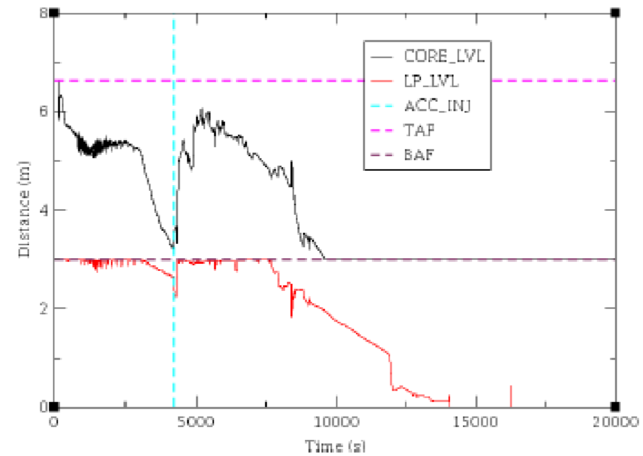
  M. Pescarini, F. Mascari, D. Mostacci, F. De Rosa, C. Lombardo, F. Giannetti,Analysis of unmitigated large break loss of coolant accidents using MELCOR code, proceedings of 35th Heat Transfer Conference, Ancona, Italy June 26-28, 2017.

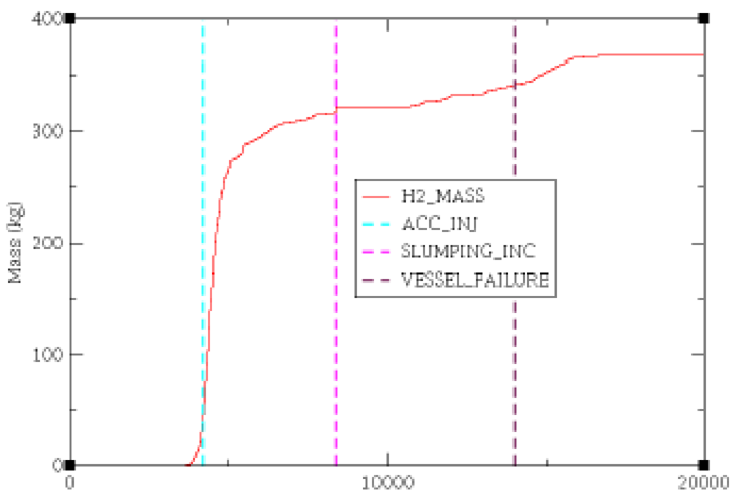
ITALIAN SAFETY ASSESSMENT ACTIVITY

- 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 conducted.
- 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. A paper about ASTEC and MAAP results has been already published:
    - **J. C. de la Rosa Blul , S. Brumm, F. Mascari, S. J. Lee, and L. Carenini, ASTEC–MAAP Comparison of a 2 Inch Cold Leg LOCA until RPV Failure, Hindawi, Science and Technology of Nuclear Installations, Volume 2018, Article ID 9189010.**
- A MELCOR analyses of a LFW and LBLOCA transient have been performed by Sapienza University of Rome. The source term evaluation is in progress.

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

- The results of the MELCOR activity have been published in:
  F. Mascari, A. Guglielmelli, J. C. de la Rosa Blul, Analysis of a postulated 2-inch Cold Leg LOCA severe accident in a PWR-like 900 MWe with MELCOR code, Proceedings of 27th International Conference Nuclear Energy For New Europe - NENE 2018, Slovenia, 10-13 September, 2018.

ITALIAN SAFETY ASSESSMENT ACTIVITY

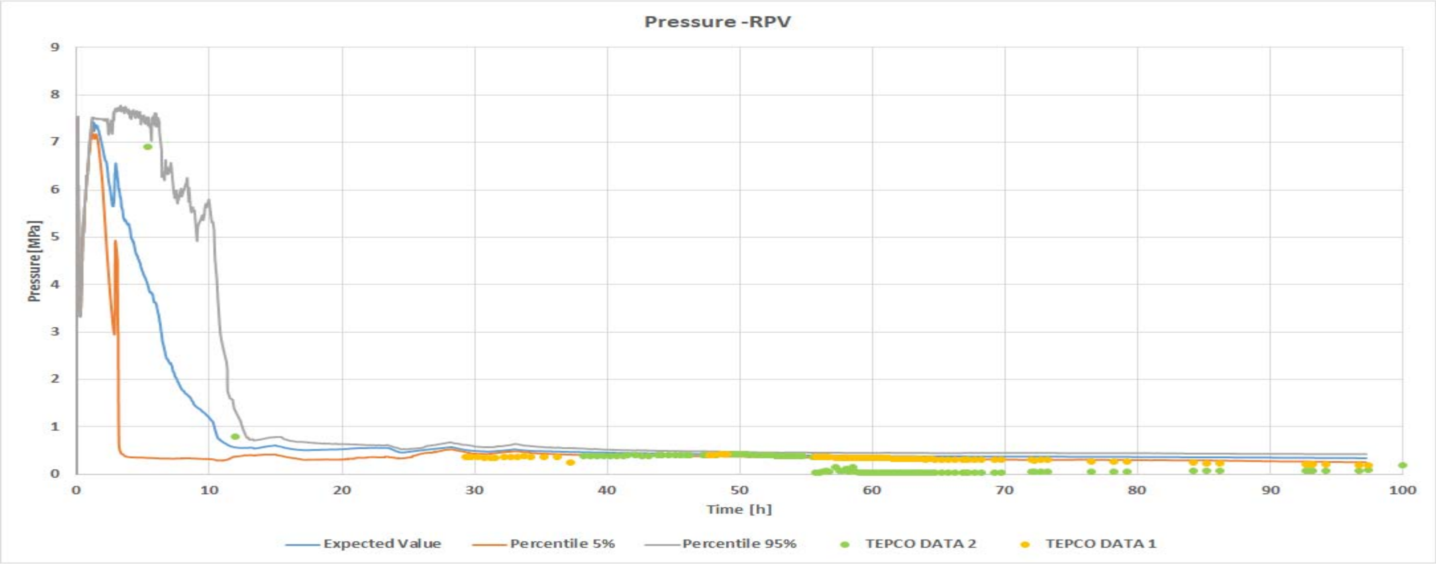
- A MELCOR analysis of the BWR FUKUSHIMA DAIICHI UNIT 1 SEVERE ACCIDENT has been performed by Sapienza University of Rome .
  - *An uncertainty analysis of the accident progression predicted by MELCOR code considering selected calculated parameters as a figure of merit has been done and presented at BEPU2018.*
  - *The uncertainty band ha been evaluated through sensitivity analyses programmed, collected and statistically manipulated through RAVEN software tool. RAVEN (**R**eactor **A**nalysis and **V**irtual control **EN**viroment) is a software tool, developed at the Idaho National Laboratory (INL), that acts as the control logic driver and post-processing tool for different applications.*
  - *MELCOR and RAVEN are internally coupled through a new Python code interface developed by Sapienza University of Rome, to perform an uncertainty quantification analysis in a core degradation transient.*
- The activity is done in collaboration with Sapienza University of Rome.

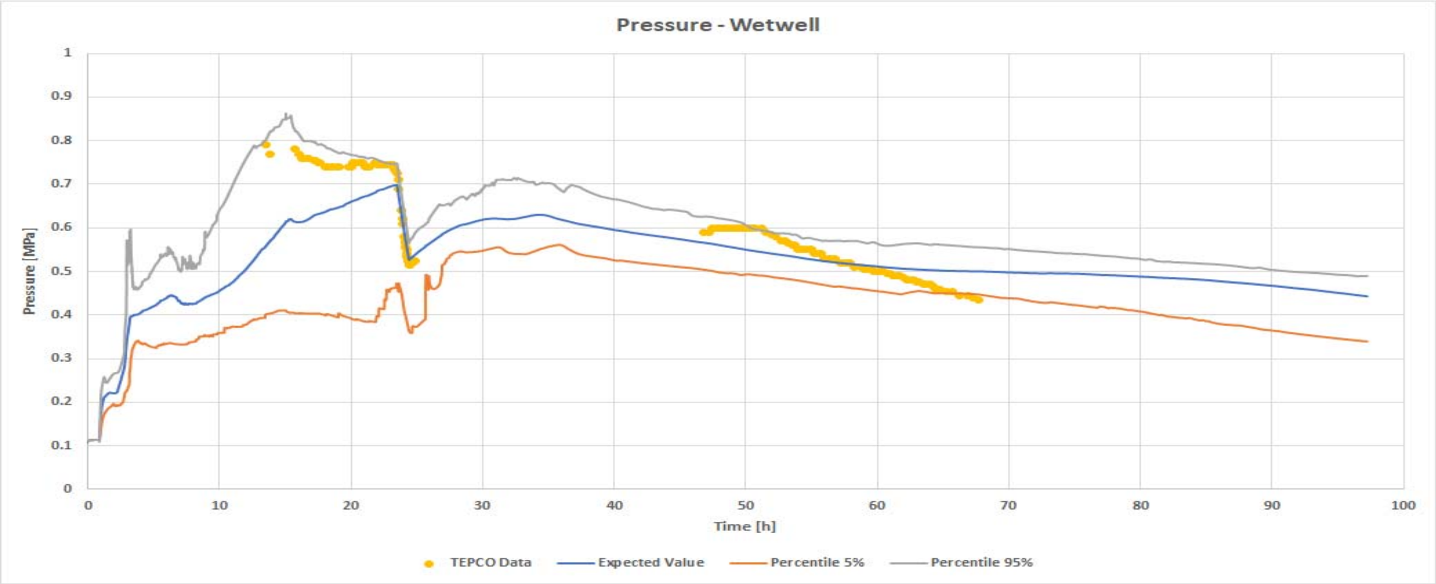
RAVEN VARIABLES AND SAMPLING FOR SENSITIVITY ANALYSIS: FUKUSHIMA DAIICHI UNIT 1 SEVERE ACCIDENT

Variable	Description	RAVEN Distribution
Area_Seal	Recirculation pump seals leak flow area	**Triangular** Mode: $9{,}2\ \mathrm{E}-5 [m^2]$ Min: $6{,}0\ \mathrm{E}-5\ [m^2]$ Max: $4{,}0\ \mathrm{E}-4\ [m^2]$
vfall	Velocity of falling debris	**Triangular** Mode: $0.1\ [m/s]$ Min: $0.05\ [m/s]$ Max: $1.2\ [m/s]$
hdblh	Heat transfer coefficient from debris to lower head	**Triangular** Mode: $1000\ [\mathrm{W}/m^2\ K]$ Min: $50\ [\mathrm{W}/m^2\ K]$ Max: $1100 [\mathrm{W}/m^2\ K]$
SC1132(1)	Core Component Failure Parameters - Temperature to which oxidized fuel rods can stand in the absence of unoxidized Zr in the cladding.	**Normal** Mean: $2700\ [K]$ Sigma: $120\ [K]$
SC1141(2)	Core Melt Breakthrough Candling Parameters - Maximum melt flow rate per unit width after breakthrough	**Triangular** Mode: 0.083 $[\mathrm{k}g/s]$ Min: 0.01 $[\mathrm{k}g/s]$ Max: 1.0 $[\mathrm{k}g/s]$
SC1502(2)	Minimum Component Masses - Minimum total mass of component subject to the maximum temperature change criterion for timestep control	**Normal** Mean: 5 $[\mathrm{k}g]$ Sigma: 1.0 $[\mathrm{k}g]$
SC1250(1)	Conduction Enhancement for Molten Components - Temperature above which enhancement is employed	**Normal** Mean: 2800 $[K]$ Sigma: 150.0 $[K]$

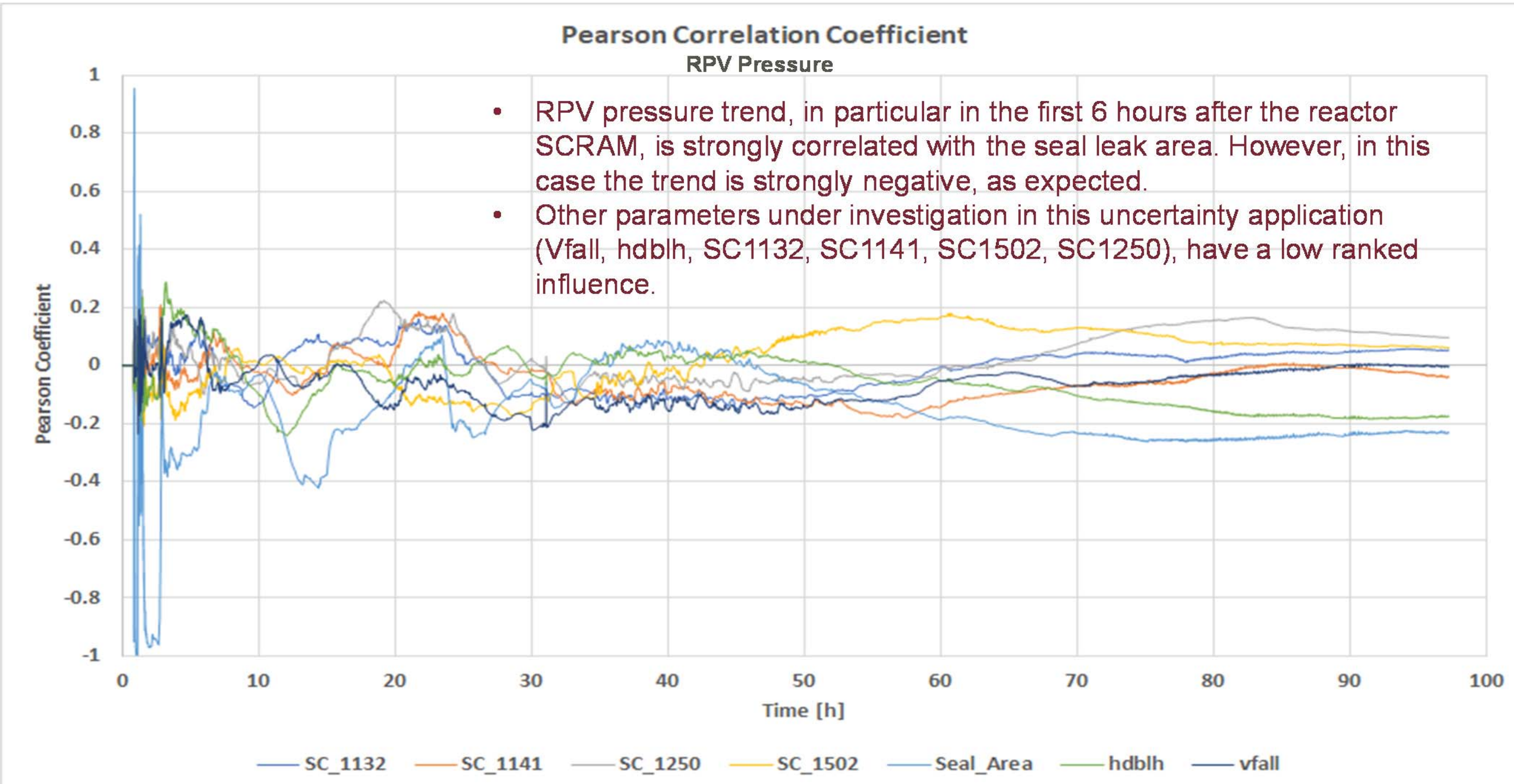
- State-of-the-Art Reactor Consequence Analyses (SOARCA) Project - Peach Bottom Integrated Analysis
- SC distribution from:Fukushima Daiichi Unit 1 Uncertainty Analysis – Exploration of Core Melt Progression Uncertain Parameters – Volume II M.R. Denman, D.M. Brooks

CODE APPLICATION AND UNCERTAINTY QUANTIFICATION: FUKUSHIMA DAIICHI UNIT 1 SEVERE ACCIDENT





CODE APPLICATION AND UNCERTAINTY QUANTIFICATION: 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.

CONCLUSIONS

- The code coupling between RAVEN and MELCOR 2.1 has been successfully implemented allowing future users to carry out sensitivity analysis and uncertainty quantification based on the variation of input parameters;

- The RAVEN code has been applied to perform a sensitivity analysis of Fukushima Daiichi unit 1, it is to underline that pressure uncertainty is mainly related to the pump seal leak area uncertainty while the MELCOR modeling parameters and sensitivity coefficients, under investigation in this application, have a low ranked influence;

- PERSPECTIVE: Selection of an extended set of parameters Perform detailed uncertainty quantification based on the TH initial conditions, boundary conditions and sensitivity coefficients of Fukushima Daichii Unit 1 and Unit 3.

- The results of the benchmark activity have been published in:

  M. D'Onorio, F. Giannetti, F. Mascari, G. Caruso, uncertainty analyses using the RAVEN software tool coupled with MELCOR severe accident code, ANS Best Estimate Plus Uncertainty International Conference (BEPU 2018) BEPU2018-282 Real Collegio, Lucca, Italy, May 13-19, 2018

ENEA

Authors thank the funding received from

Italian Minister of Economic Development

LIQUID METAL REACTOR ACTIVITY: ENEA CONTRIBUTION TO THE EU-JASMIN PROJECT

LIQUID METAL REACTOR ACTIVITY: ENEA CONTRIBUTION TO THE EU-JASMIN PROJECT

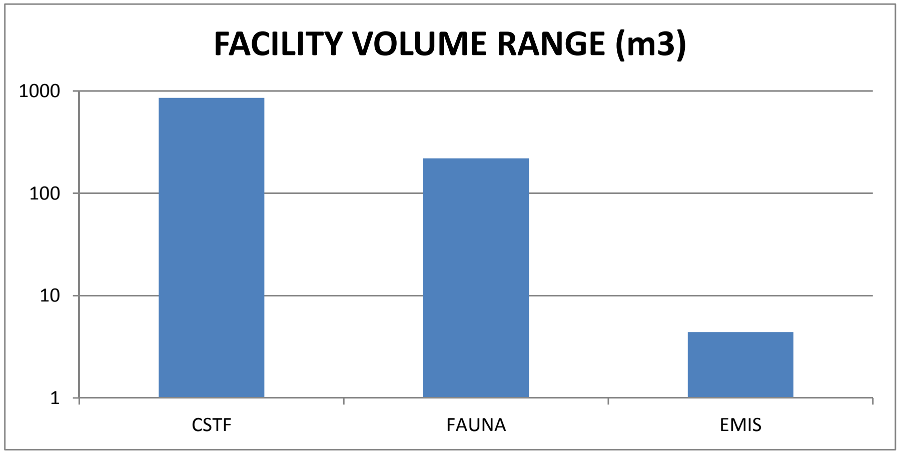
- In the framework of the European Project JASMIN - Joint Advanced Severe Accidents Modelling and Integration for Sodium-Cooled Fast Neutron Reactors-, coordinated by IRSN, ENEA has been involved in the WP2.3 (ST) - Source term - coordinated by CIEMAT.

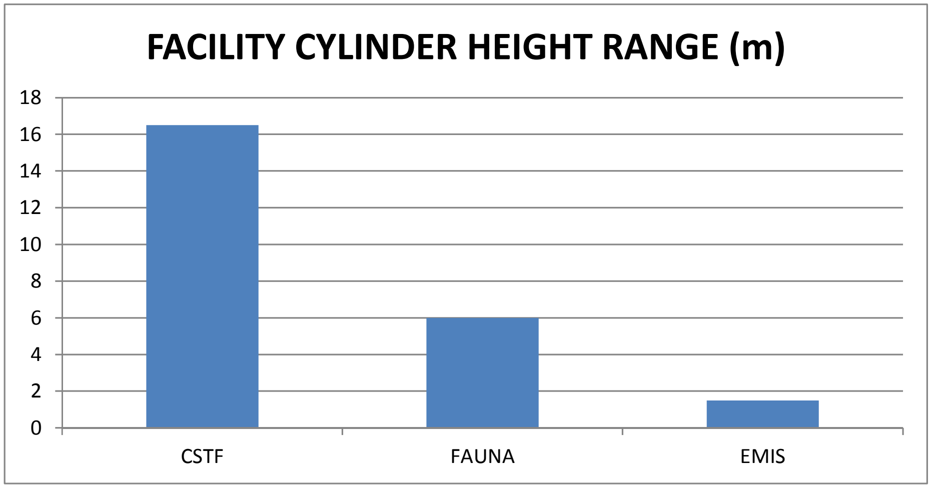
- A benchmark activity, coordinated by CIEMAT, has been performed; the involved codes are MELCOR, ASTEC-CPA, CONTAIN, FEUMIX, ASTEC-CPA*(specific models for in-containment Na phenomena have been implemented). The tests selected for the benchmark are the CSTF-AB1, AB2, and FAUNA F2, F3 and EMIS10b(Pool Fire Tests).

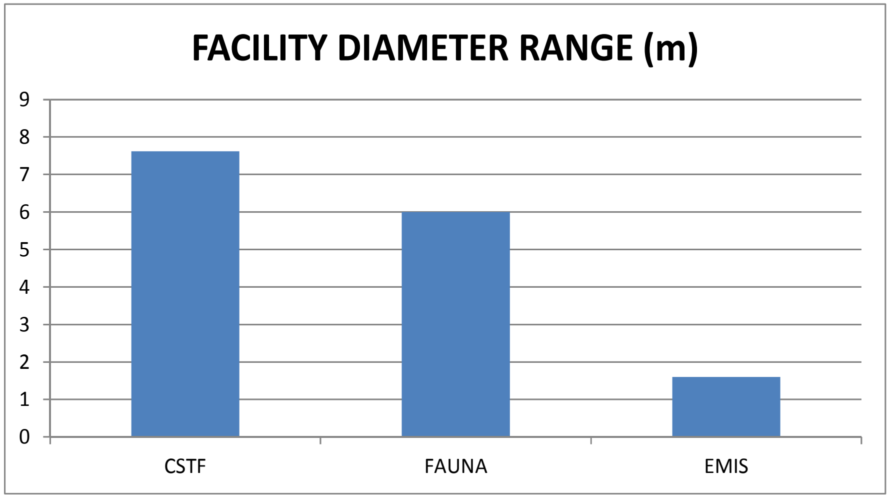
- ENEA used the MELCOR and ASTEC-CPA code for the simulation of the selected test in order to support the benchmark of the CPA module of ASTEC-NA.

LIQUID METAL REACTOR ACTIVITY FRAMEWORK GEOMETRY DIMESIONS INVESTIGATED

PARAMETERS	CSTF	FAUNA	EMIS
GEOMETRY			
Type	Cylindrical	Cylindrical	Cylindrical
Diameter (m)	7.62	6	1.6
Cylinder Height (m)	16.5	6	1.5
Volume (m3)	852	220	4.4
Surface area for heat transfer			
Na Pool(m2)	4.42 (F2) or 12 (F3) (H)		0.125 (H)
Top Head (m2)	63	31.5 (H)	2.32 (H)
Bottom Head(m2)	63	31.5 (H)	2.32 (H)
Cylinder (m2)	394	113 (V)	7.54 (H)
Tot Vessel Shell (m2)	520	176	12.3015
THICKNESS FOR HEAT TRANSFER			
Wall Thickness (mm)	18.1 (TH and BH) 22.9 (CYL)	16	-







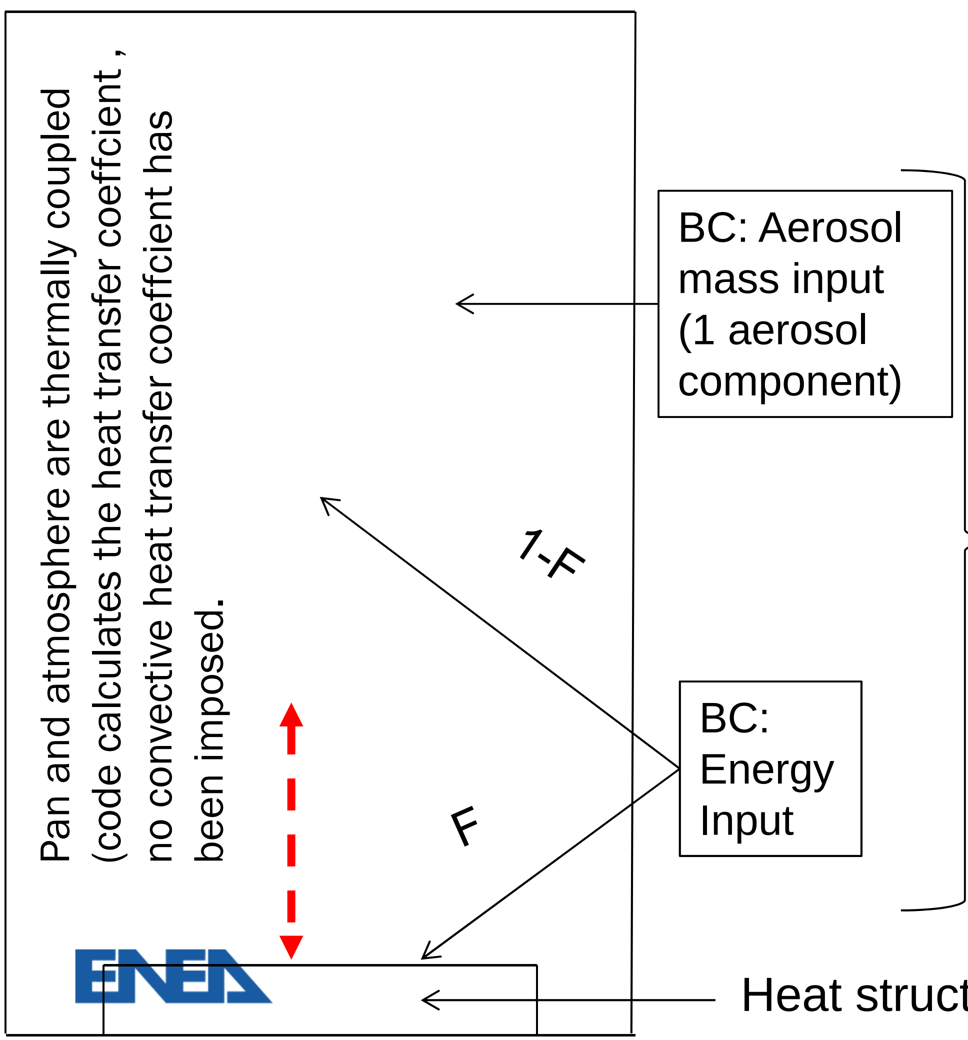
LIQUID METAL REACTOR ACTIVITY FRAMEWORK TEST MATRIX INVESTIGATED

PARAMETER	CSTF-AB1	CSTF-AB2	FAUNA-F2	FAUNA F3	EMIS10b
Initial containment atmosphere					
Oxygen concentration (%)	19.8	20.9	17-25	15-25*	20
Temperature (K)	299.65	293.65	298.15	298.15	294.05
Pressure (MPa)	0.125	0.128	0.101	0.101	0.1013
Dew point (K)	283.15	280.75	-	-	-
Steam addition					
Flow started (seconds after t0)	-	960	-	-	-
Flow stopped (seconds after t0)	-	4560	-	-	-
Flow rate (kg/s)	0	0.019	0	0	0
Sodium spill					
Sodium mass delivered (kg)	410	472	250	500	9.4
Initial sodium temperature (K)	873.15	873.15	773.15	773.15	545
Pool fire burning area (m2)	4.4	4.4	2	12	0.125
Pool fire burn duration (s)	3600	3600	12600	4800	approx 6000
Aerosol source					
Type	Na pool fire	Na pool fire	Na pool fire	Na pool fire	Na pool fire
Total sodium oxidized (kg)	156.5	174.8	170	460	-
Average oxidation rate (kg Na/h·m2)	35.7	39.9	-	33	-
Total aerosol mass released (kg Na)	39.9	38.6	-	-	-
Average aerosol release rate (kg Na/h·m2)	9.11	8.81	-	9	-
Average aerosol mass source (kg Na/s)	0.0111	0.0107	-	-	-
Fraction of oxidized Na release as aerosol	0.255	0.221	-	-	-
Source duration (s)	3600	3600	12600	4800	6000

APPROACH USED FOR THE SIMULATION

ONE VOLUME APPROACH

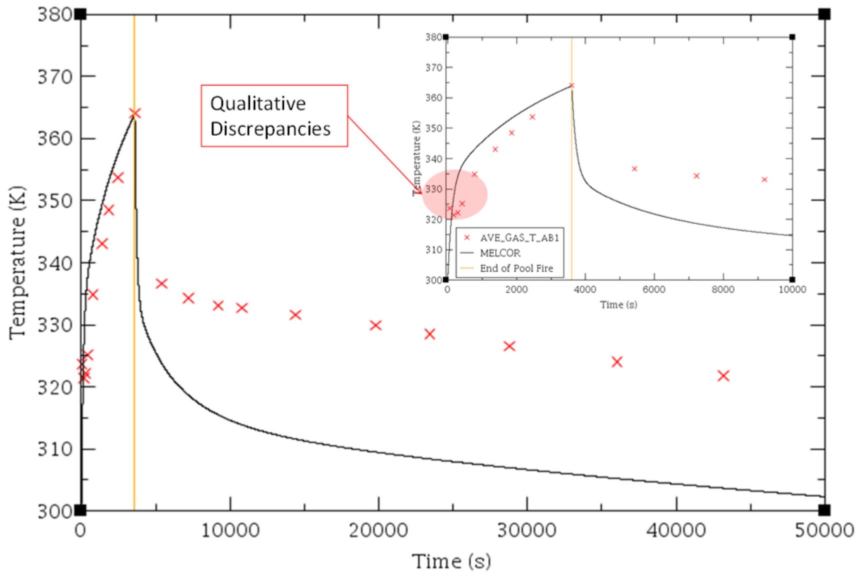
- Recirculation mass flow is not simulated;
- Temperature distribution is uniform;
- Aerosol concentration is uniform;
- Heat transfer phenomena are simulated (Radiation is considered).

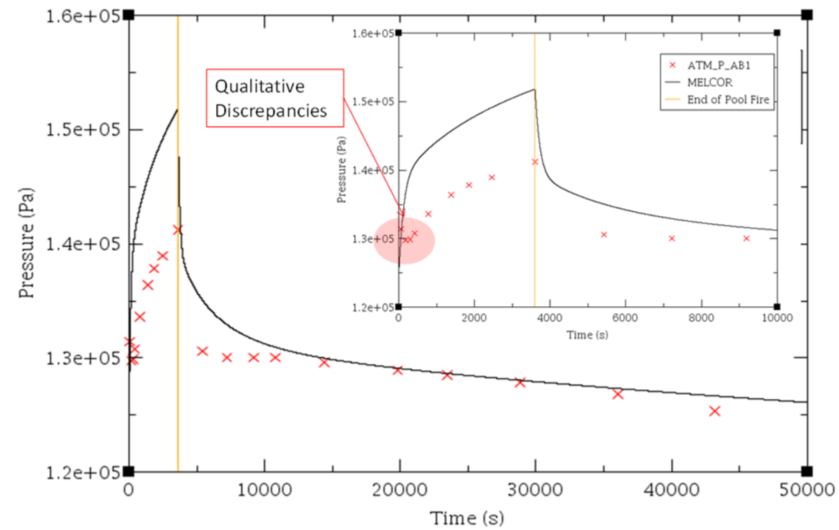


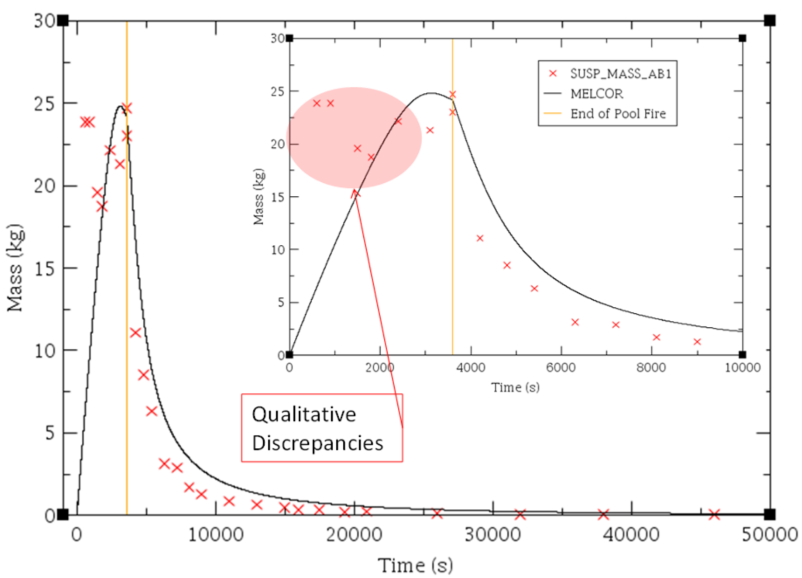
- No chemical reaction are modelled;
  - No flame region is modelled;
  - No diffusion phenomena are considered;
- Only one aerosol species is considered as user input (MELCOR Aerosol Class 2- Alkali Metals-Li, Na, K, Rb, Cs, Fr, Cu)
- The combustion energy realease is modeled as user input

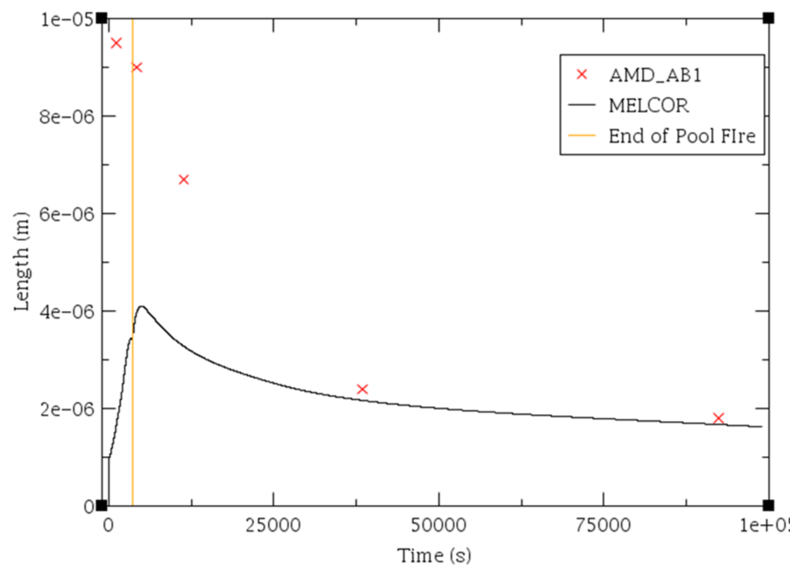
Heat structure made of Sodium

CODE APPLICATION: MAIN EXPERIMENTAL OBSERVATIONS AGAINST MELCOR CALCULATED DATA DURING THE AB1 TEST

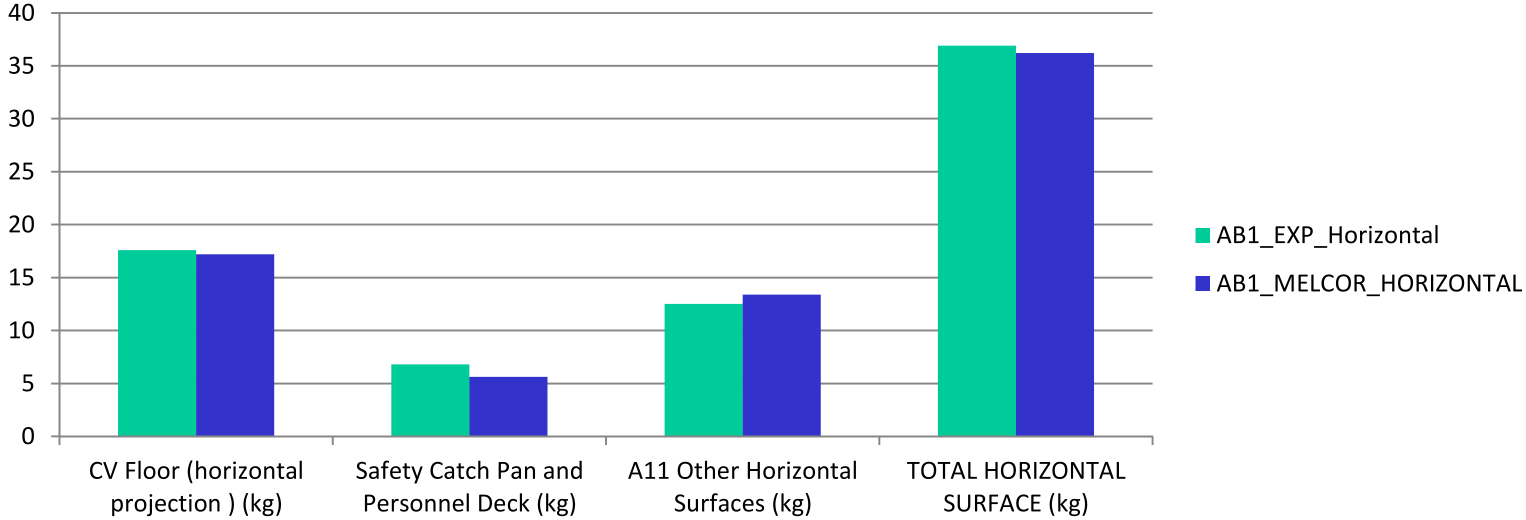


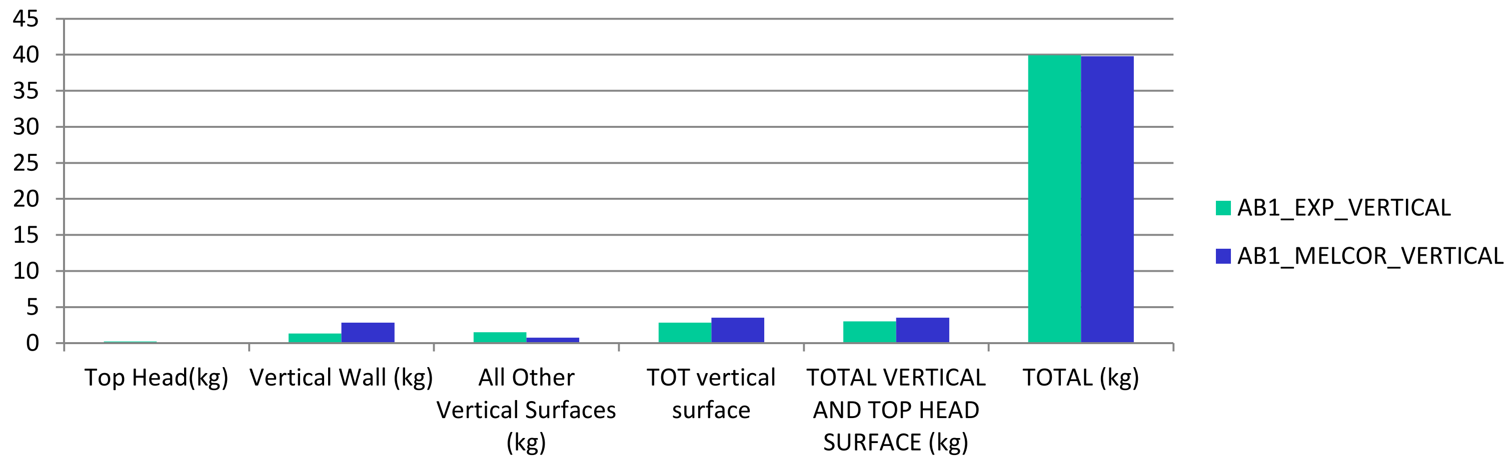






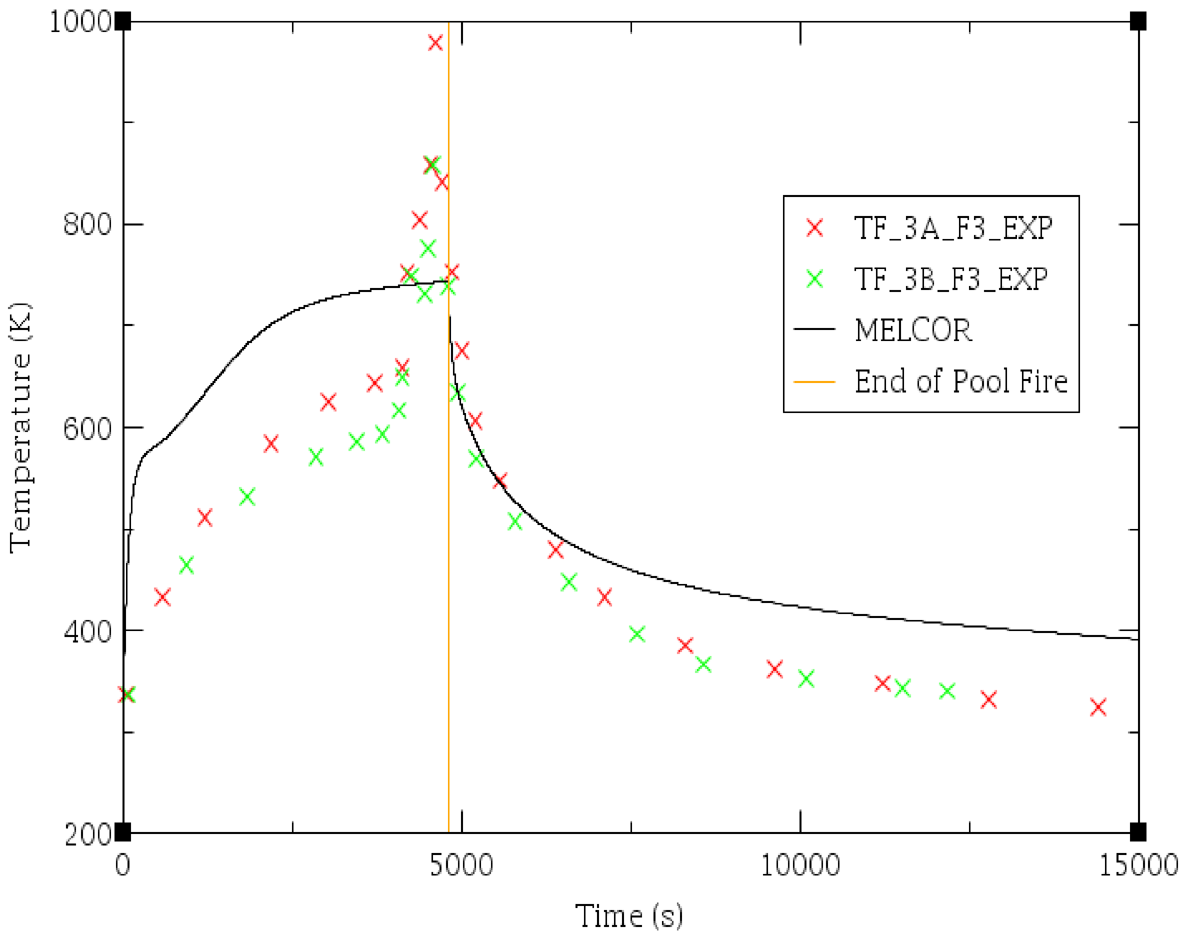
CODE APPLICATION: MAIN EXPERIMENTAL OBSERVATIONS AGAINST MELCOR CALCULATED DATA DURING THE AB1 TEST





The code, by using default aerosol constants value, show a quantitative close prediction of the experimental data

CODE APPLICATION: MAIN EXPERIMENTAL OBSERVATIONS AGAINST MELCOR CALCULATED DATA DURING THE F3 TEST

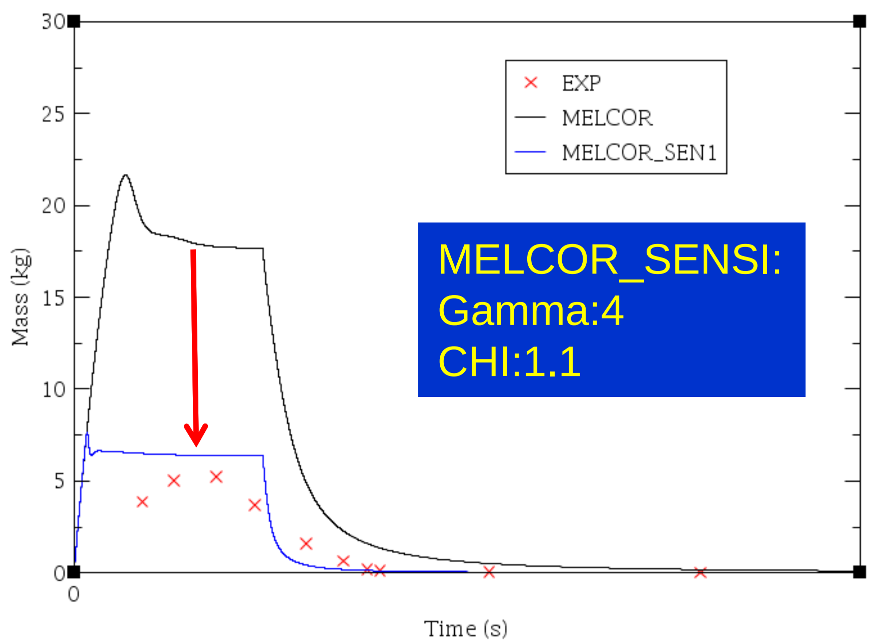


- Here two local temperature are considered. The thermocouples are
  - At the same axial position: at about 4 m above the burn pan
  - Two diferent radial positions.

- The comparison between experimental and calculated data shows, as expected, that the ''one cell approach'' is not suitable to quantitatively simulate containment behavior when not uniform condition are present in the facility.

Warning: the experimental data, reported in the plot, are obtained directly from existing figure, therefore there is an error associated with the withdrawal process.

CODE APPLICATION: MAIN EXPERIMENTAL OBSERVATIONS AGAINST MELCOR CALCULATED DATA DURING THE F3 TEST



- The total suspended mass behavior is qualitatively predicted by the code.
- After stopping the aerosol release, in agreement with the experimental data, the code predicts a reasonable decrease of the tot suspended mass.
- MELCOR code show a general quantitative over prediction of this parameter by using the default aerosol coefficient

- From the experimental data related to the F3 test it is obtained that the Dynamic shape factor is 1.1.
- Based on the analyses performed with PARDISEKO code a CHI:1.1 and gamma =4 is used *[Cherdron, W., Jordan S., 1985].*
- This last calculation show a better quantitative agreement of the MELCOR calculated data with the experimental data
- MELCOR Analyses are consistent with previous code calculation available in the scientifica literature

CODE APPLICATION: GENERAL MAIN EXPERIMENTAL OBSERVATIONS AGAINST MELCOR CALCULATED DATA

- In general the code are able to qualitatively reproduce the expected main phenomena of interest.
- As expected *the single cell approach is reasonable for the simulation of the thermal hydraulic containment behavior when uniform condition are achieved;*
- More detailed thermal hydraulic nodalization is required when not well mixing condition inside the facility are reached;
- A general over prediction of the pressure behavior is observed;
- Codes is, in general, able to predict the particle size time evolution, though a general underestimation of the AMMD is in general obtained by the code.
- The qualitative behavior of the suspended mass is in general predicted by the code. Sensitivity analyses show a general agreement of the MELCOR code with previous analyses showing the importance of the GAMMA and shape factor.
- It is in general to underline that MELCOR code shows a reasonable prediction of the mass fractions deposited distribution by using its default aerosol constant value.
- The results of the benchmark activity have been published in:
  - L.E. Herranz, M. Garcia, L. Lebel, F. Mascari, C. Spengler, In-containment source term predictability of ASTEC-Na: Major insights from data-predictions benchmarking, Nuclear Engineering and Design 320 (2017) 269–281

Authors thank the funding received from the 7th Framework Programme of the European Commission via the JASMIN project;

Special thanks for IRSN for the coordination of Project;

Special thanks for CIEMAT for the coordination of the WP2.3 activities

FIRST EXERCISES MELCOR/DAKOTA COUPLING:

IMPORTANT:

<u>These activities represents only the first ENEA 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</u>

FIRST EXERCISE MELCOR DAKOTA COUPLING:

UNCERTAINTY ANALYSES FIRST EXERCISE : MELCOR/DAKOTA IN A SNAP ENVIROMENT/ARCHITECTURE

- Starting from the AB1 analyses done with MELCOR code a simple excercise has been conducted only to test the DAKOTA software tool coupled with MELCOR code in SNAP environment/architecture.
- Considering previous analyses, the qualitative behavior of the suspended mass is in general predicted by the code. Sensitivity analyses show the importance of the shape factors (GAMMA and CHI).
- The target of this exercise is not to investigate the dependence of MELCOR results by the aerosol constant, but considering this dependence test the DAKOTA/MELCOR coupling in a SNAP environment/architecture. The analyses is not exhaustive and the attention is mostly focused on aerosol suspended mass.

AEROSOL CONSTANT	Range Used	MELCOR NAME	Default Value MELCOR
Dynamic Shape factor	1.0-5.0	CHI	1
Agglomeration Shape factor	1.0–5.0	GAMMA	1
Slip Coefficient	1.14-1.257	FSLIP	1.257
Sticking Coefficient	0.5-1.0	STICK	1
Turbulence Dissipation (m2/s3)	0.001-0.02	TURBDS	1.00E-03
Thermal Accomodation Coeffcient	2.18-2.25	FTHERM	2.25
Gas Thermal Conductivity/Particle Thermal Conductivity	0.06	TKGOP	0.05
Diffusion Boundary layer Thickness (m)	1.0e-5-1.0e-3	DELDIF	1.00E-05

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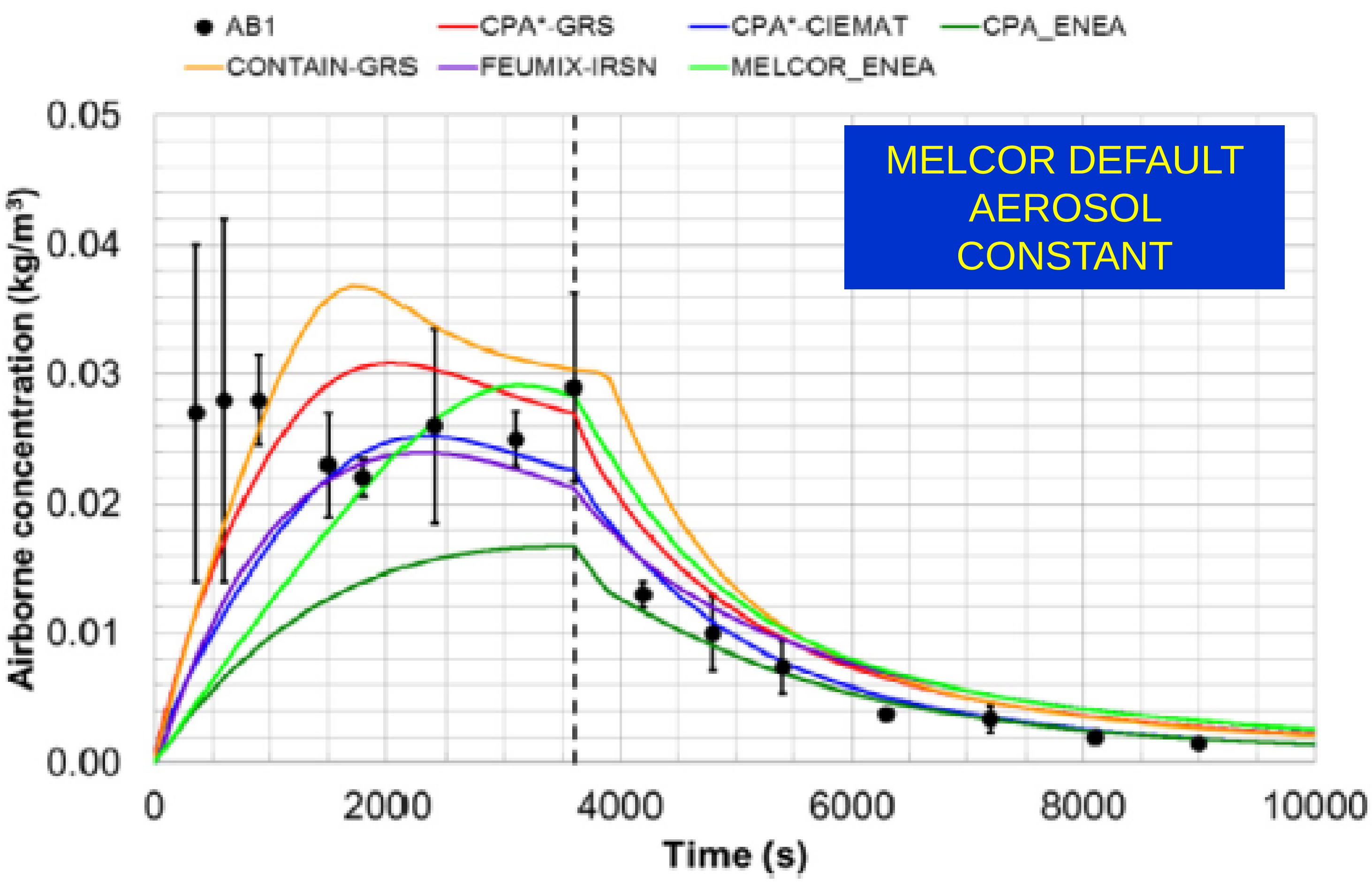
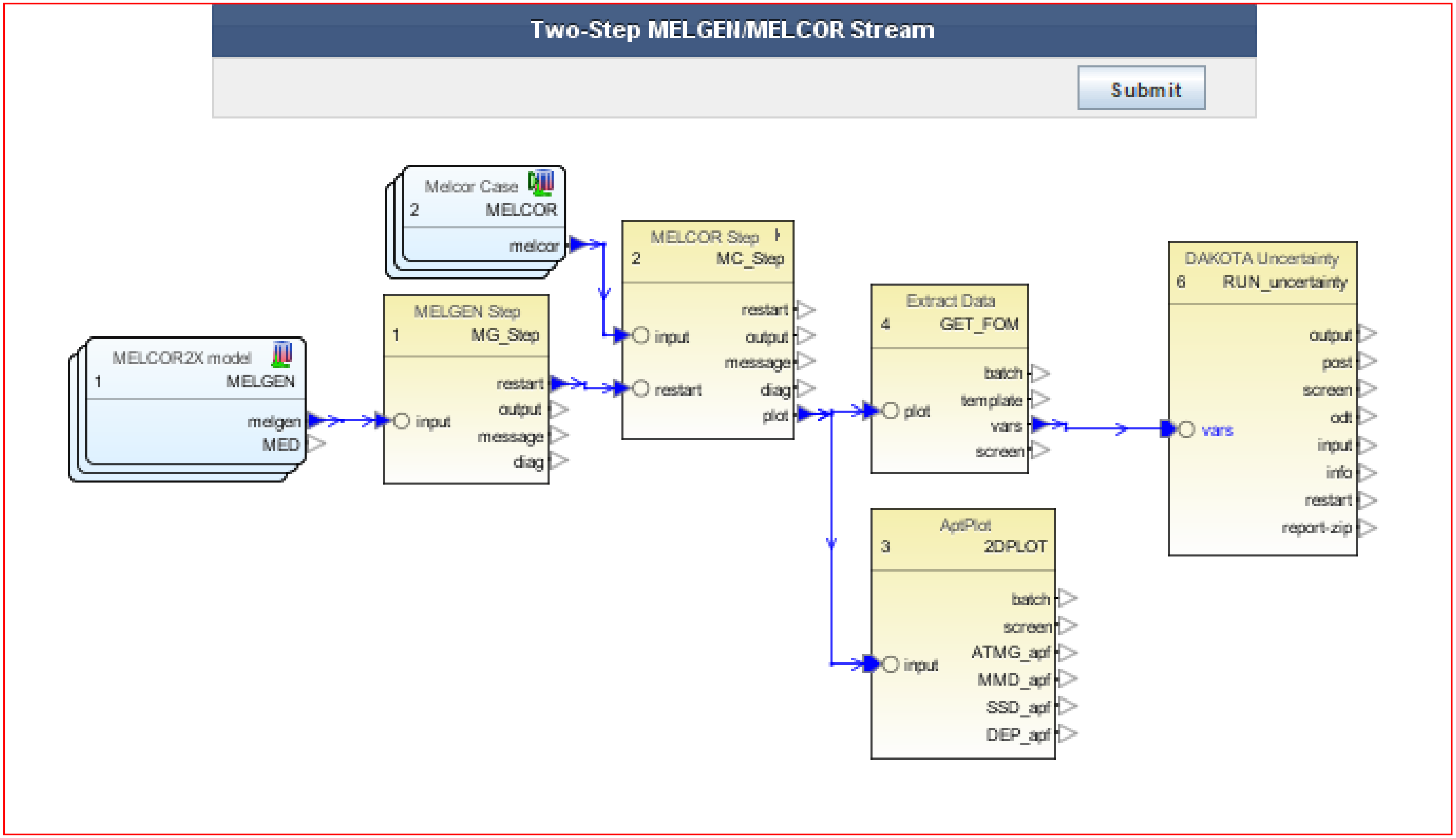
COMPARISON DONE, WITH THE PROJECT PARTNERS, DURING THE EU- JASMIN PROJECT BENCHMARK RELATED TO AIRBORNE CONCENTRATION



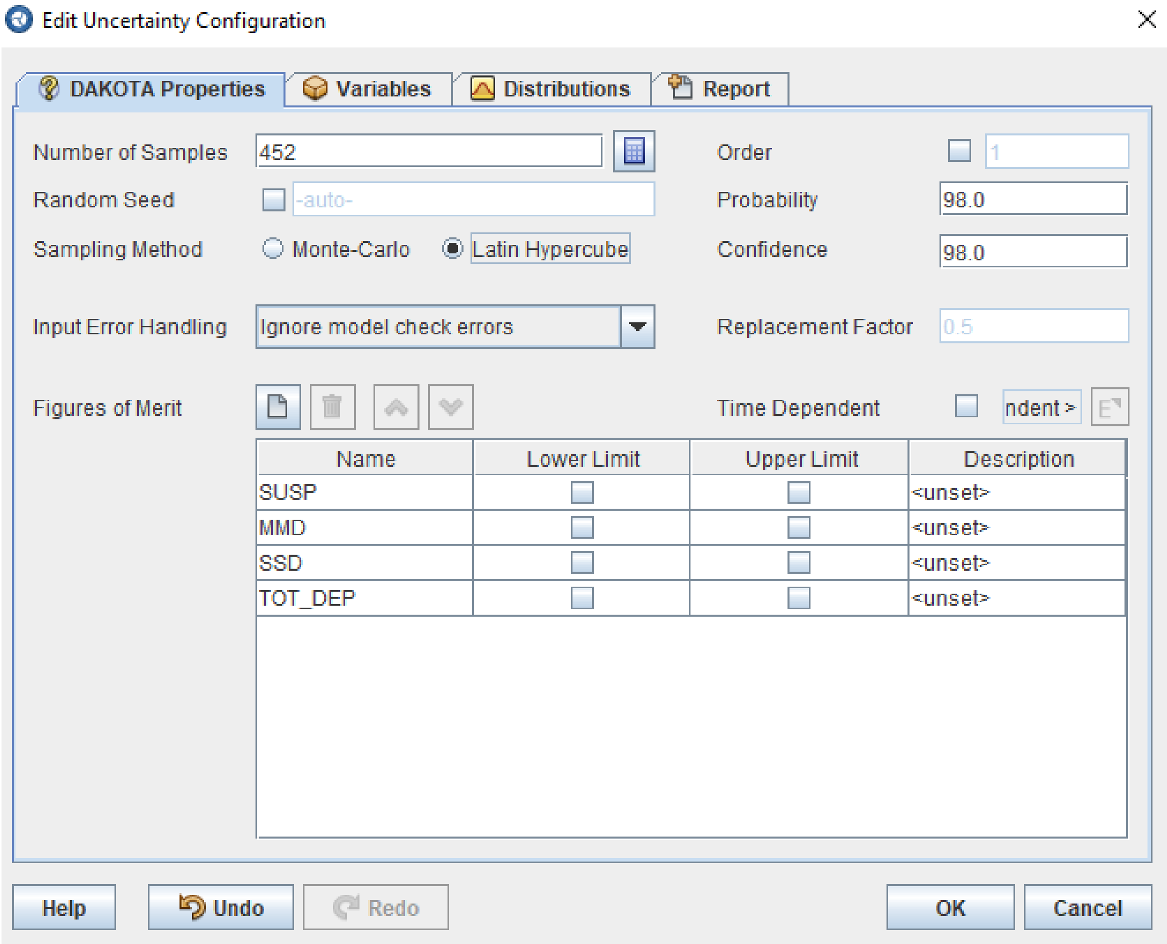
L.E. Herranz, M. Garcia, L. Lebel, F. Mascari, C. Spengler, In-containment source term predictability of ASTEC-Na: Major insights from data-predictions benchmarking, Nuclear Engineering and Design 320 (2017) 269–281

UNCERTAINTY ANALYSES: MELCOR/DAKOTA IN A SNAP ENVIROMENT/ARCHITECTURE

DAKOTA UNCERTAINTY SNAP STREAM



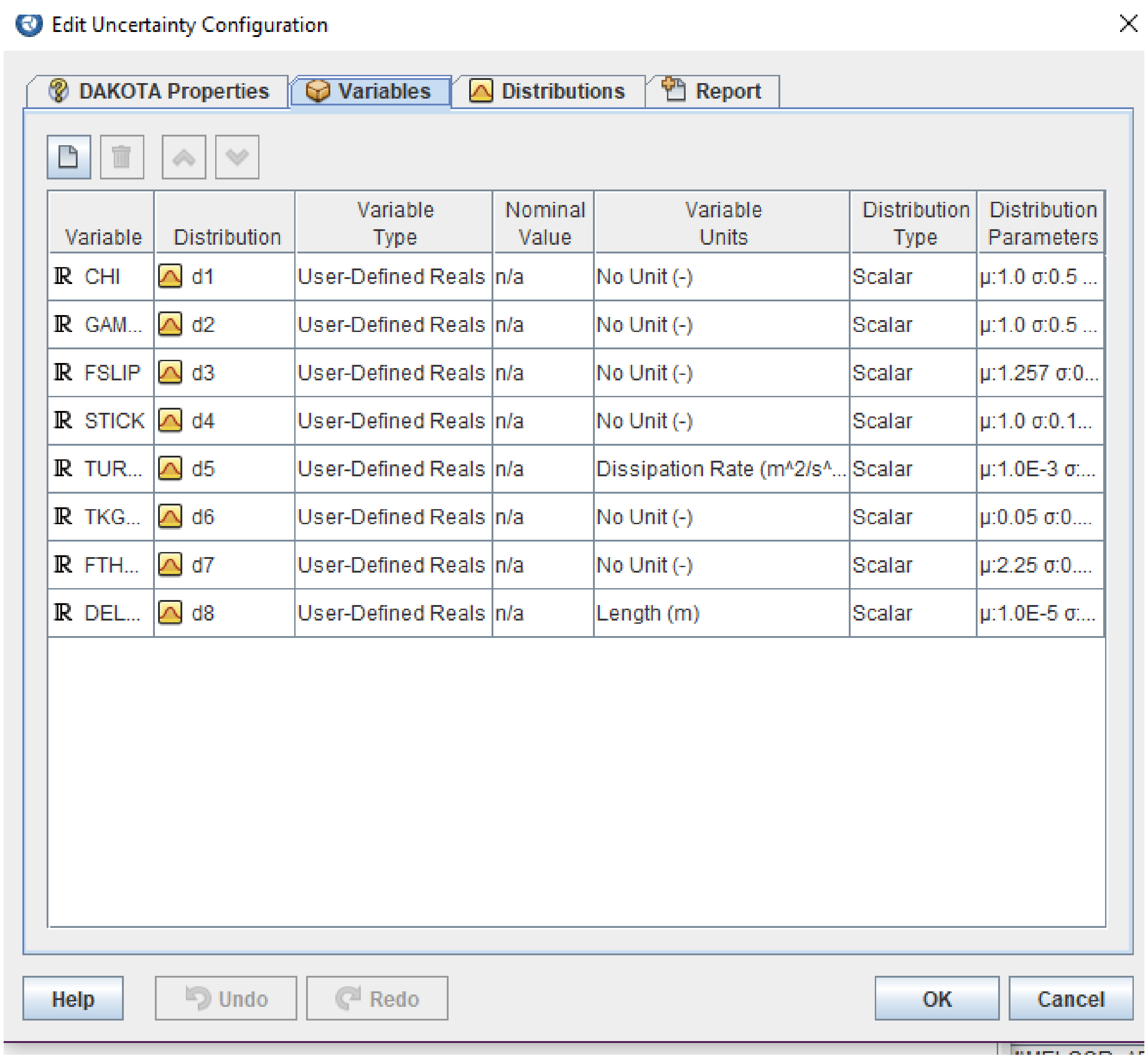
UNCERTAINTY ANALYSES: FIGURE OF MERIT DEFINITIONS



The following four figures of merit are defined for this uncertainty analysis:

- SUSP: Suspended airborne concentration
- MMD: Aerosol mass median diameter
- SSD: Geometric standard deviation of the aerosol distribution
- TOT_DEP: Tot mass deposited

MODEL VARIABLES AND DISTRIBUTIONS DEFINITION



FIRST TENTATIVE DEFINITION ONLY TO TEST THE PROCEDURE

UNCERTAINTY ANALYSES APPLICATION

- An initial DAKOTA run was performed using the specified input parameters to generate a set of variates for each task. The individual tasks were then performed and the figures of merit were extracted from the completed calculations.

- A total of 452 tasks are required to calculate the 0 order statistic for the specified FOMs with a 98.0% probability and a 98.0% confidence level. 452 tasks were completed successfully.

- As example the data extracted to perform the uncertainty analyses is the value of the FOMS at the end of the pool fires.

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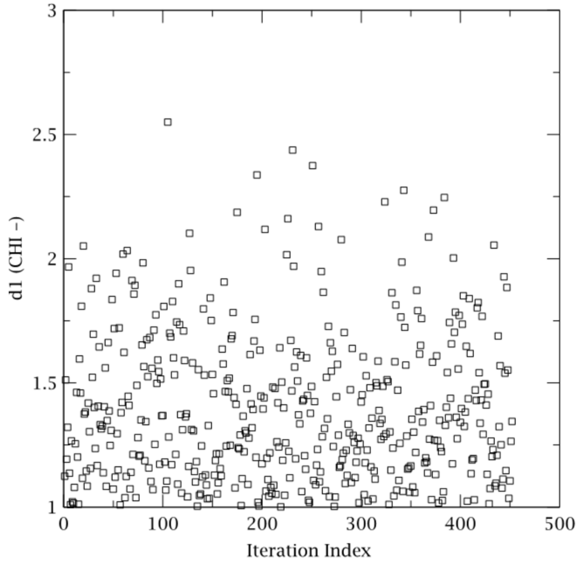
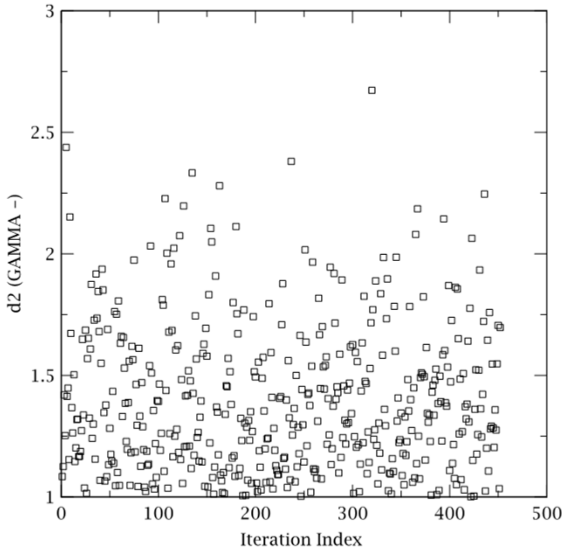
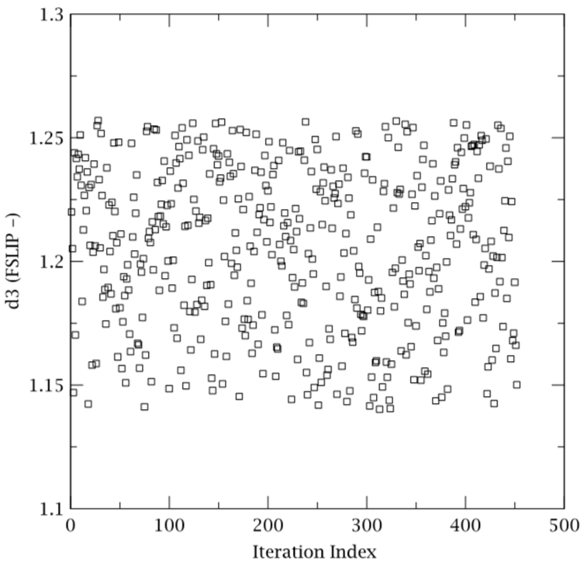
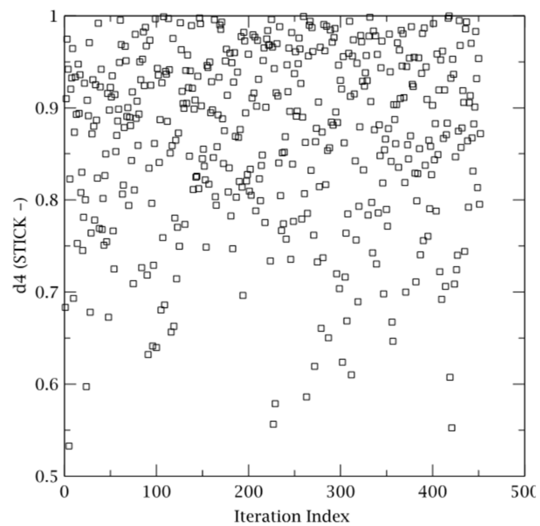
VARIATE AND RESPONSE DATA: MODEL VARIABLES VS ITERATION

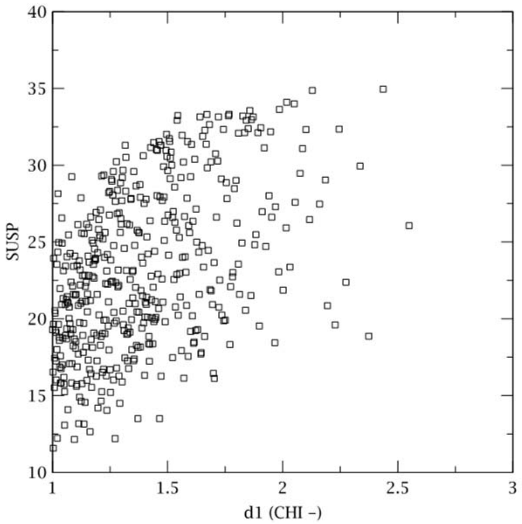


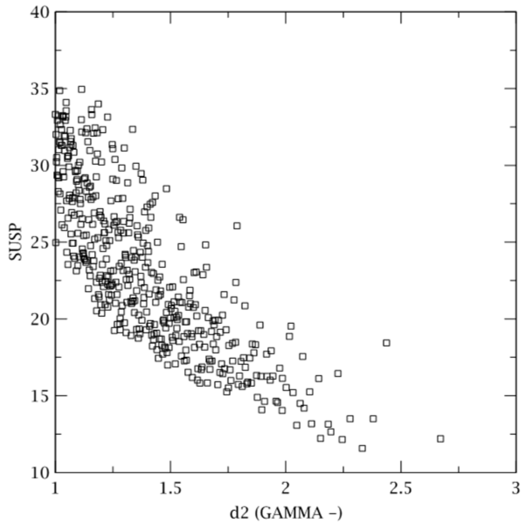


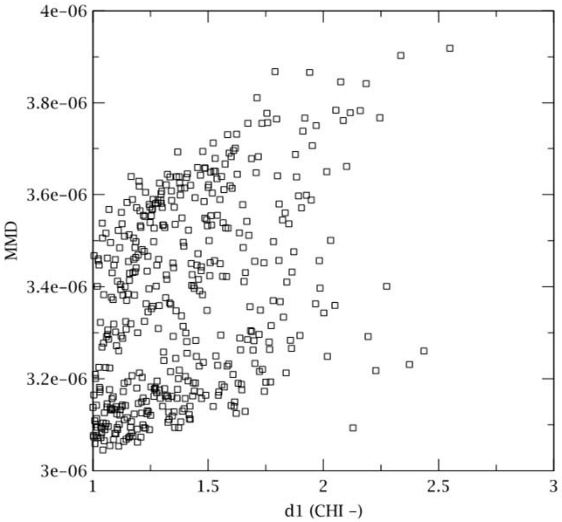




VARIATE AND RESPONSE DATA: FOM VS MODEL VARIABLE







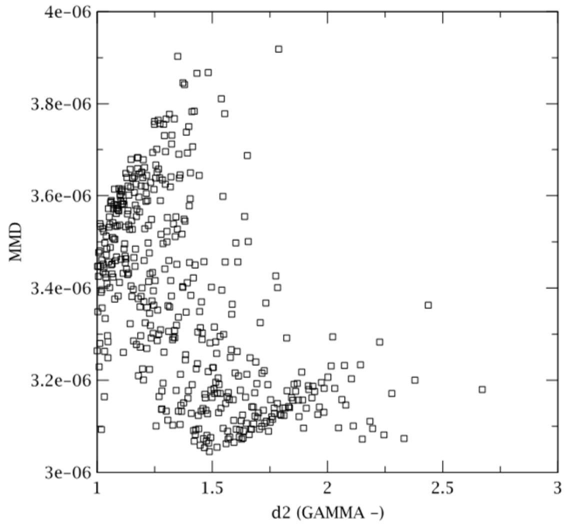
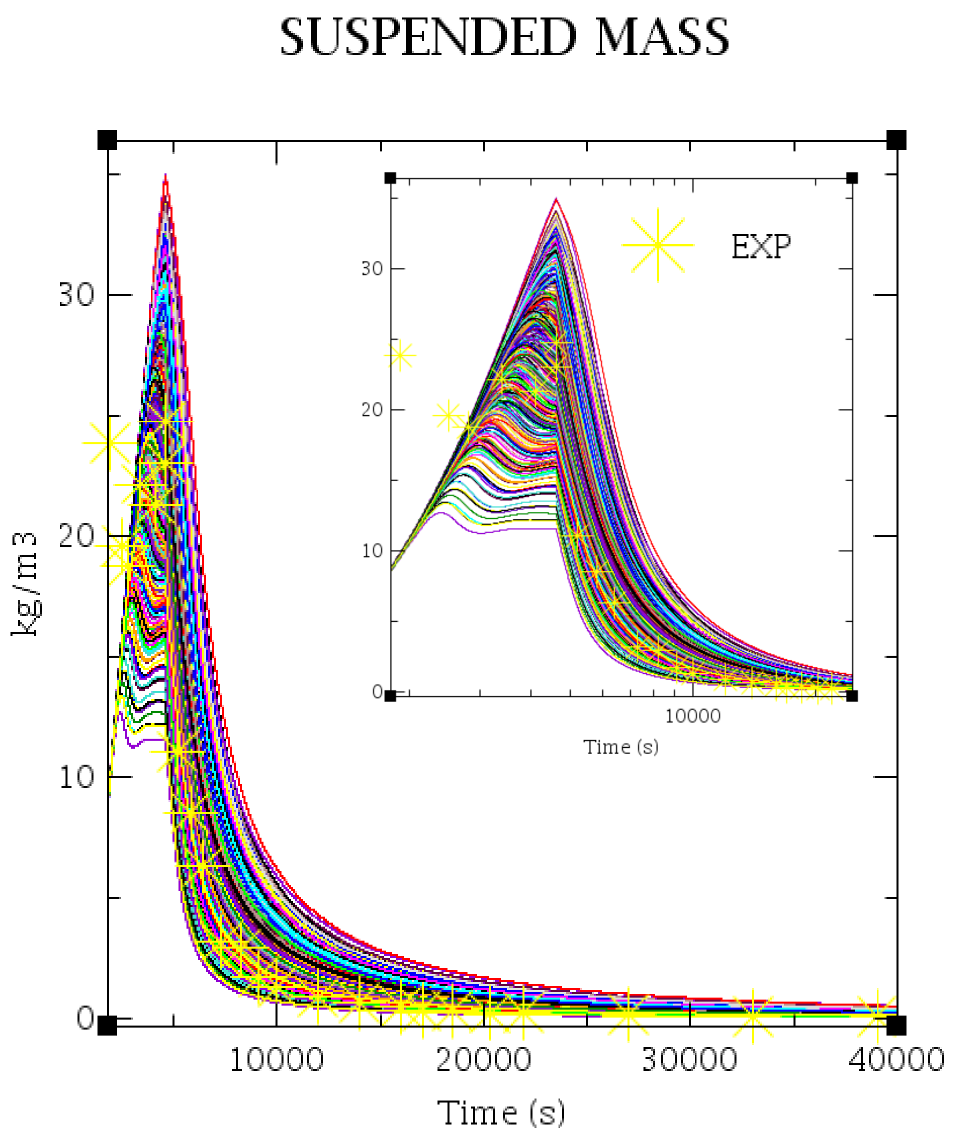
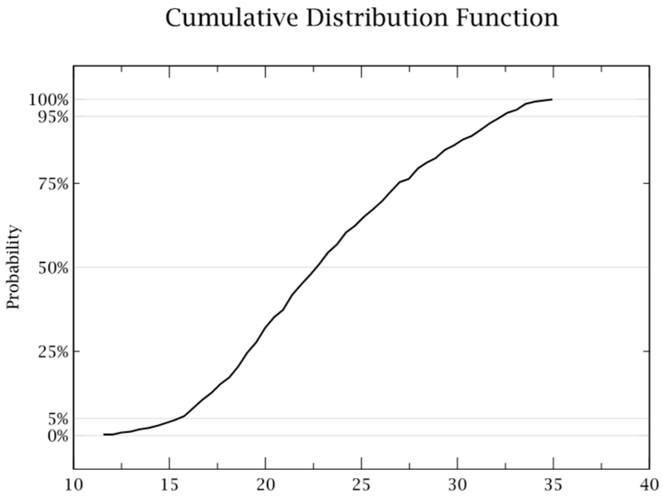
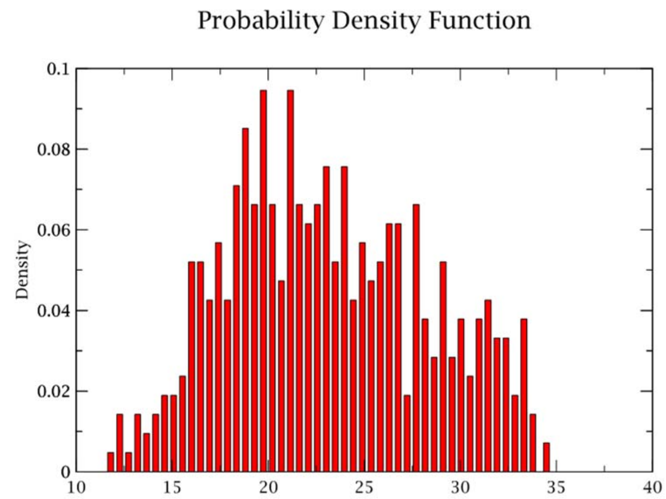


FIGURE OF MERIT APTPLOT AND DAKOTA RESULTS







DAKOTA RESULTS FOR THE SUSPENDED AIRBORNE CONCENTRATION

Statistical results based on 452 samples:

Summary	Value	Task #
Min Value	11,5619	135
Max Value	34,94785	231
Mean	23,1486	-
Median	22,61615	average of 185 and 102
Standard Deviation	5,24126	-
Coefficient of Variance	0,20899	-

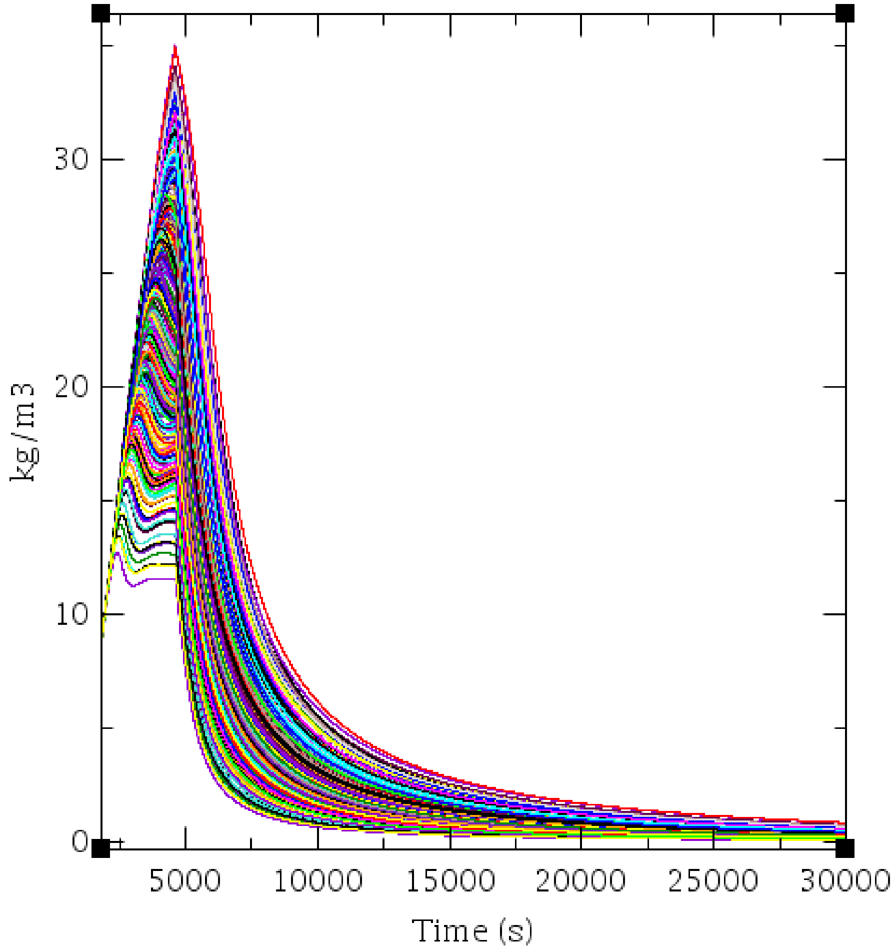
Response Correlations

	Simple	Partial	Simple Rank	Partial Rank
d1	0.470256	0.87473	0.467447	0.91701
d2	-0.802127	-0.949568	-0.834265	-0.971159
d3	0.0224959	0.058378	0.0222934	0.0260806
d4	-0.168458	-0.604287	-0.180916	-0.639637
d5	-0.0273861	0.00756861	-0.0303348	-0.211284
d6	0.0183224	0.00143064	0.0099558	-0.0508664
d7	-0.00344123	0.0352195	0.00241676	-0.0439138
d8	0.0159181	-0.0395132	0.0320765	0.0407834
MMD	0.750518	-	0.820846	-
SSD	-0.683356	-	-0.604208	-
TOT_DEP	-1.0	-	-1.0	-

Note: NaN values typically indicate an insufficient number of tasks were supplied to perfom the analysis.

FIGURE OF MERIT APTPLOT AND DAKOTA RESULTS

SUSPENDED MASS



DEPOSITED MASS

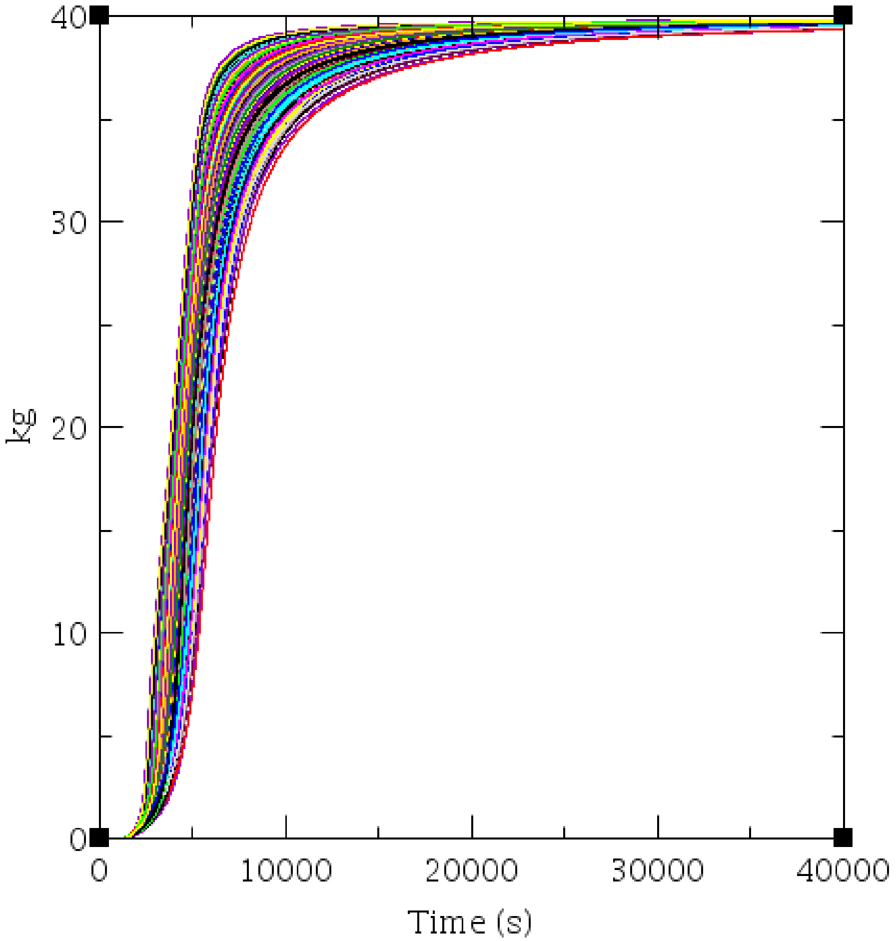
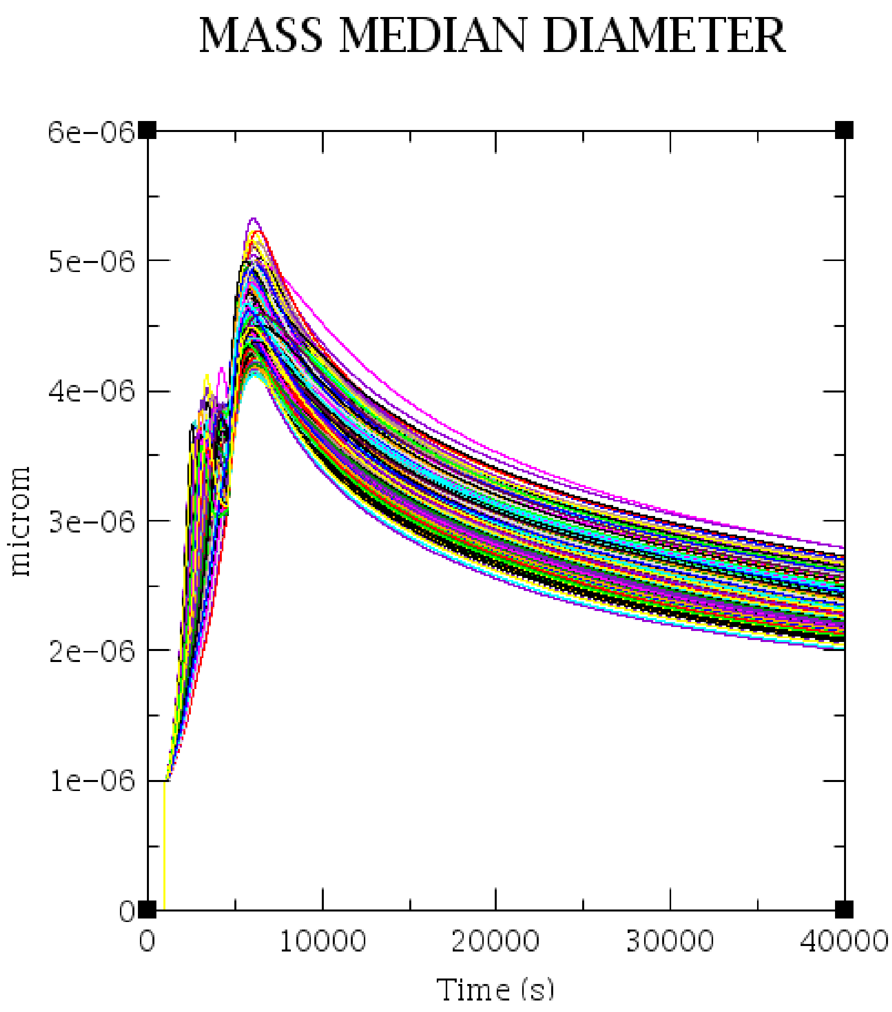
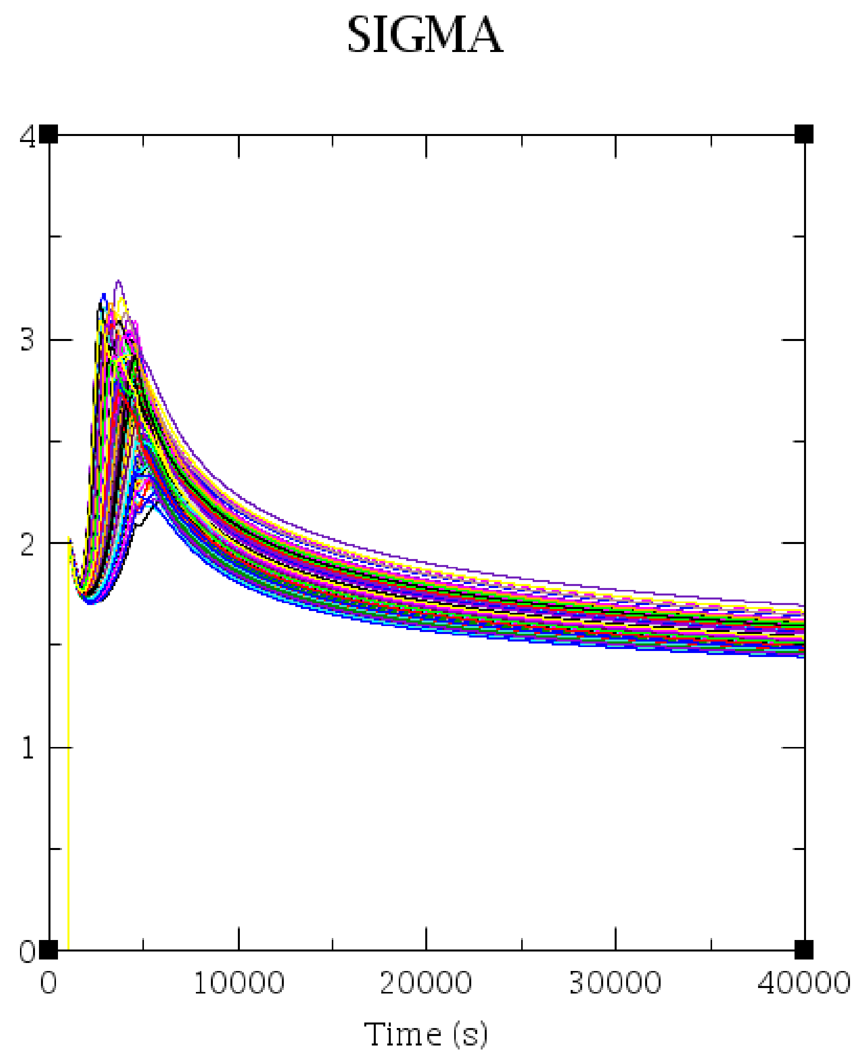


FIGURE OF MERIT APTPLOT AND DAKOTA RESULTS





SECOND EXERCISE MELCOR DAKOTA/COUPLING

MELCOR/DAKOTA COUPLING PWR EXAMPLE

- In the CSARP framework, a MELCOR/DAKOTA sample input-deck has been developed by ENEA using the _PWR_v2-0.inp input available for MELCOR user;

- The steps that have been done are:
  - Get the text file : _PWR_v2-0.inp;
  - Imported through SNAP and create a .med file;
  - Create the STREAM MELCOR and DAKOTA;
  - Choose some uncertainty input parameters and the related distribution Type and parameters
  - to run the analyses with SNAP;

- As a figure merit of this first analysis, only the hydrogen generation at the end the transient (COR-DMH2-TOT) has been selected.

MELCOR/DAKOTA COUPLING PWR EXAMPLE

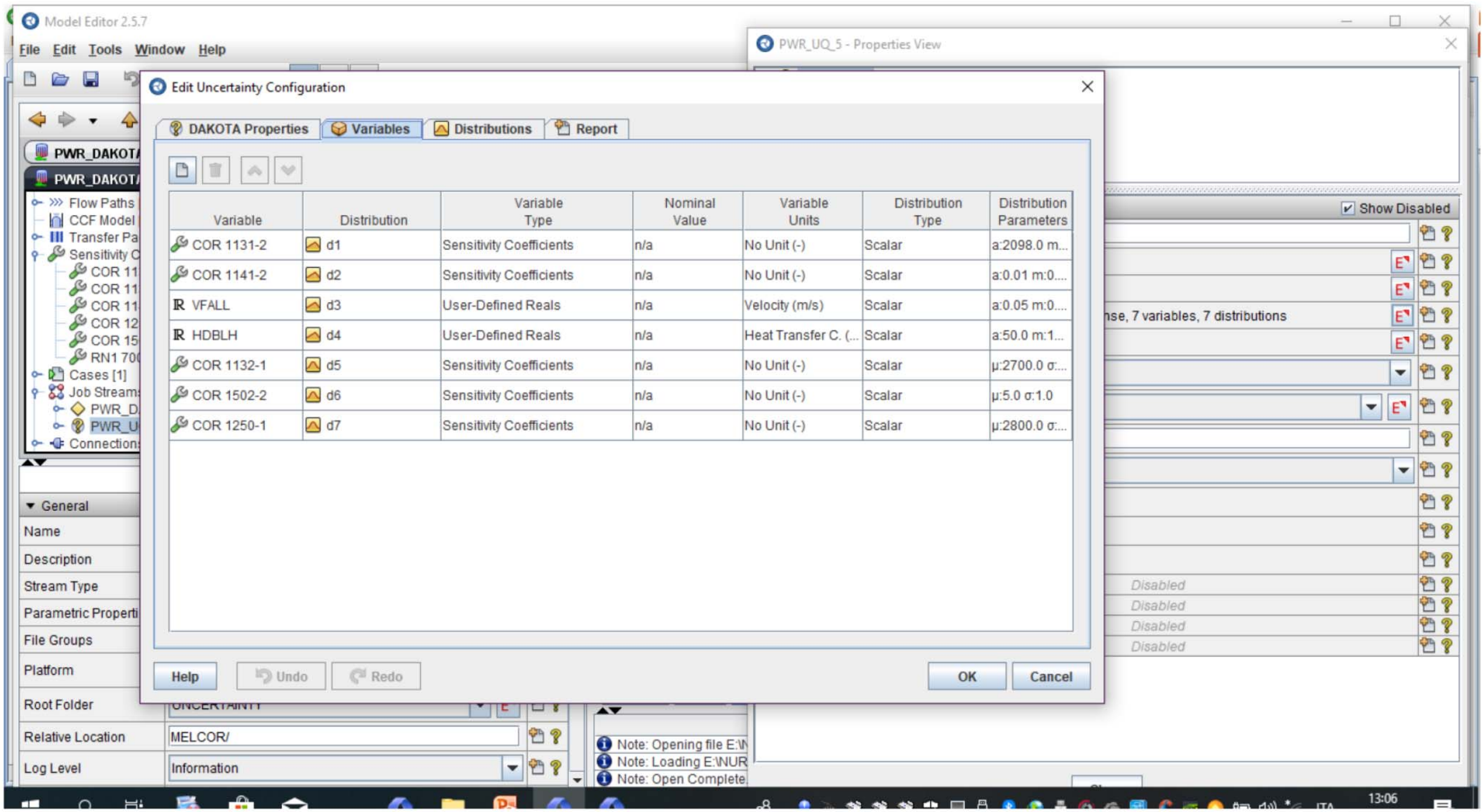
- The uncertainty parameters that have been selected to show the procedure, as examples, are:
  - *Vfall : Velocity of falling debris*
  - *Hdblh: Heat transfer coefficient from debris to lower head*
  - *SC1132(1): Core Component Failure Parameters - Temperature to which oxidized fuel rods can stand in the absence of unoxidized Zr in the cladding.*
  - *SC1131(2): Zircaloy melt breakout temperature*
  - *SC1141(2): Core Melt Breakthrough Candling Parameters - Maximum melt flow rate per unit width after breakthrough*
  - *SC1502(2): Minimum Component Masses - Minimum total mass of component subject to the maximum temperature change criterion for timestep control*
  - *SC1250(1): Conduction Enhancement for Molten Components - Temperature above which enhancement is employed*

- These parameters have been selected only as example to show all the procedure and a first tentative parameter distributions has been used.

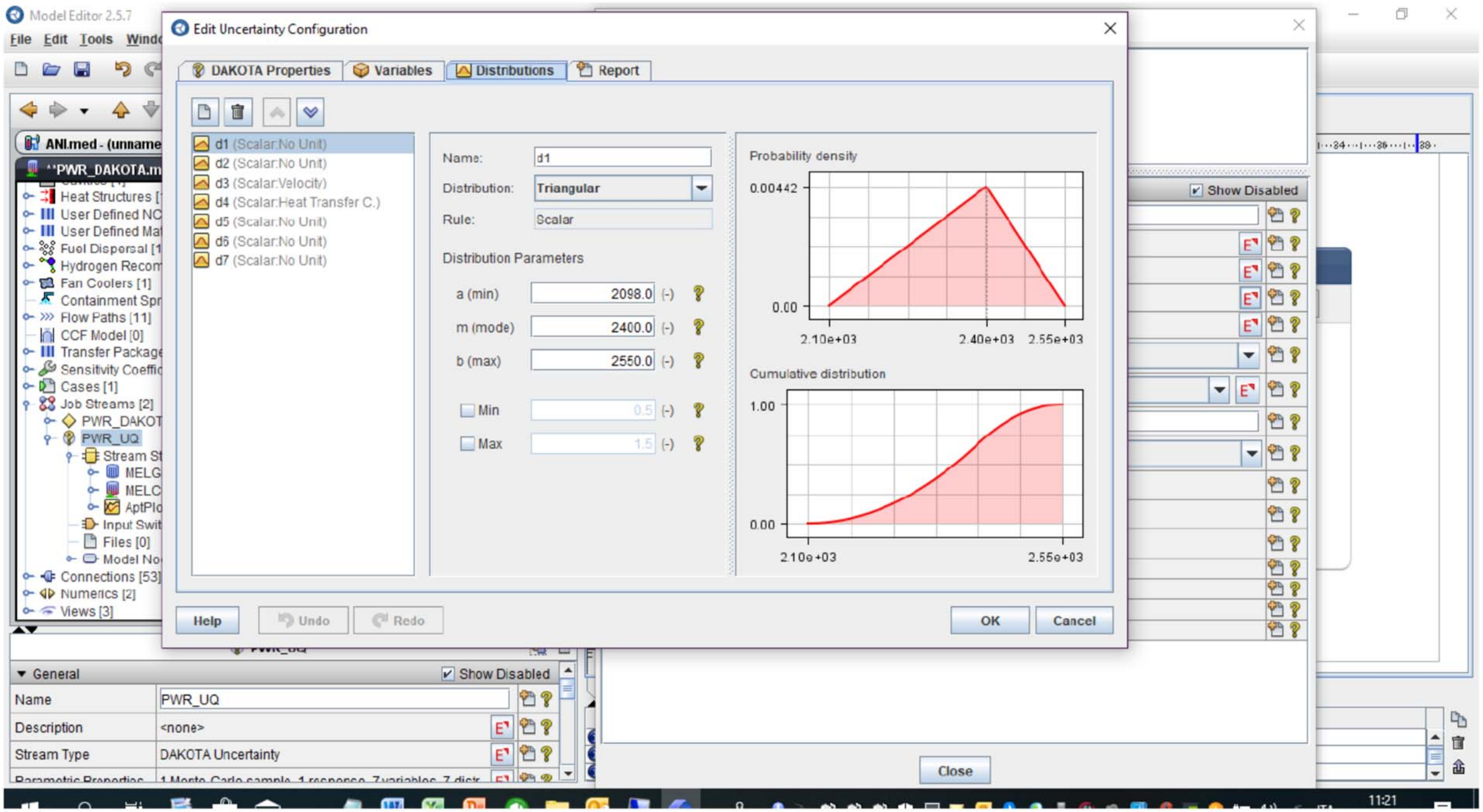
INPUT UNCERTAIN PARAMETERS SELECTED FOR THE ANALYSIS AND FIRST TENTATIVE PARAMETER DISTRIBUTIONS

Parameter	Distribution TYpe	Distribution Parameter
*Vfall : Velocity of falling debris [m/s]**	Triangular	Min: 0.05 Mode: 0.1 Max: 1.2
*Hdblh: Heat transfer coefficient from debris to lower head [W/m2h]**	Triangular	Min:50 Mode:1000 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. [K]	Normal	MEAN: 2700 SDTV: 120
SC1131(2): Zircaloy melt breakout temperature	Triangular	Min: 2098 Mode: 2400 Max: 2550
*SC1141(2): Core Melt Breakthrough Candling Parameters - Maximum melt flow rate per unit width after breakthrough [kg/s]**	Triangular	Min: 0.01 Mode: 0.083 Max: 1
*SC1502(2): Minimum Component Masses - Minimum total mass of component subject to the maximum temperature change criterion for timestep control [kg]**	Normal	MEAN: 5 STDV: 1
*SC1250(1): Conduction Enhancement for Molten Components - Temperature above which enhancement is employed [K]**	Normal	MEAN: 2800 STDV: 150

DAKOTA UNCERTAINTY STREAM- EDIT UNCERTAINTY CONFIGURATION- VARIABLES



DAKOTA UNCERTAINTY STREAM- EDIT UNCERTAINTY CONFIGURATION-DISTRIBUTION



JOB-STATUS DURING DAKOTA UNCERTAINTY APPLICATION

calcserv://Local/UNCERTAINTY/MELCOR/PWR_UQ/

Job	Priority	Job Type	Status	Submitted	Started	Completed	Calc Time	Loaded	Evaluation
PWR_UQ	4	Stream	Complete	Aug 05 12:21	Aug 05 12:21	Aug 05 12:33	No Data	No	
2DPLOT	5	AptPlot	Complete	Aug 05 12:30	Aug 05 12:32	Aug 05 12:32	No Data	No	
GET_FOM_T01	5	AptPlotExtract	Complete	Aug 05 12:24	Aug 05 12:30	Aug 05 12:30	No Data	No	
GET_FOM_T02	5	AptPlotExtract	Complete	Aug 05 12:24	Aug 05 12:30	Aug 05 12:30	No Data	No	
GET_FOM_T03	5	AptPlotExtract	Complete	Aug 05 12:24	Aug 05 12:30	Aug 05 12:30	No Data	No	
GET_FOM_T04	5	AptPlotExtract	Complete	Aug 05 12:24	Aug 05 12:30	Aug 05 12:30	No Data	No	
GET_FOM_T05	5	AptPlotExtract	Complete	Aug 05 12:25	Aug 05 12:30	Aug 05 12:31	No Data	No	
GET_FOM_T06	5	AptPlotExtract	Complete	Aug 05 12:25	Aug 05 12:30	Aug 05 12:31	No Data	No	
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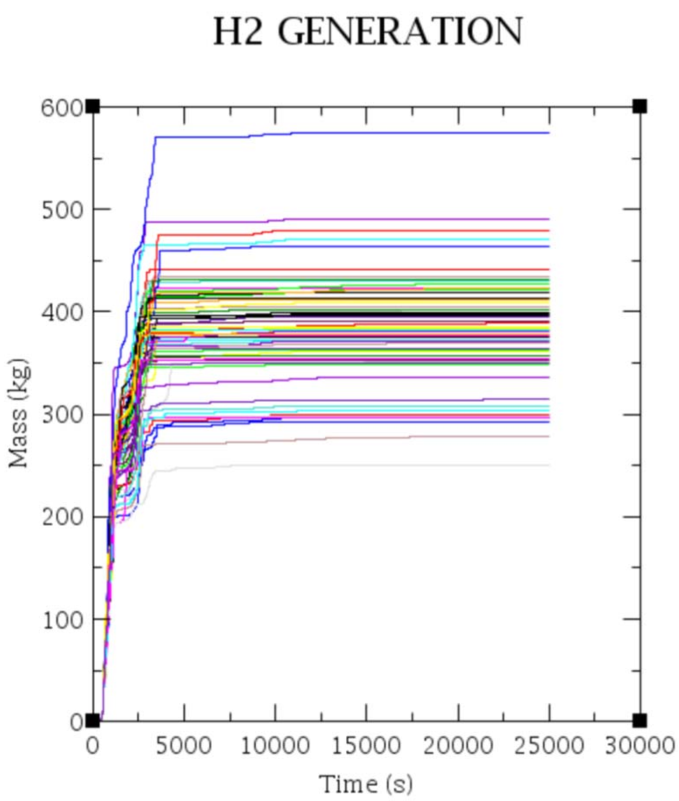
GET_FOM_T32	5	AptPlotExtract	Complete	Aug 05 12:27	Aug 05 12:31	Aug 05 12:31	No Data	No	
GET_FOM_T33	5	AptPlotExtract	Complete	Aug 05 12:27	Aug 05 12:31	Aug 05 12:31	No Data	No	

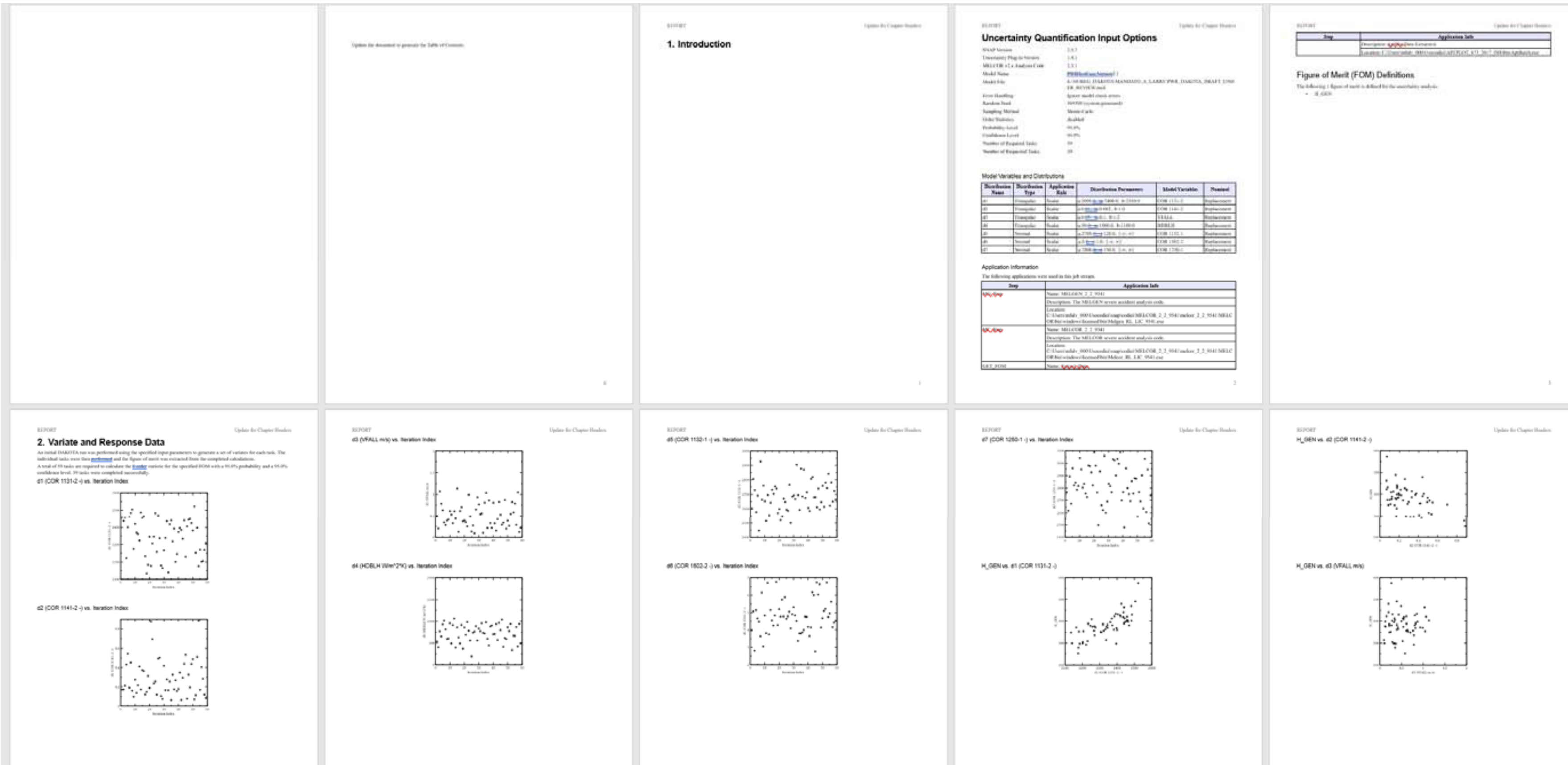
DISPERSION OF THE TOT HYDROGEN MASS GENERATED AND RESPONSE CORRELATION



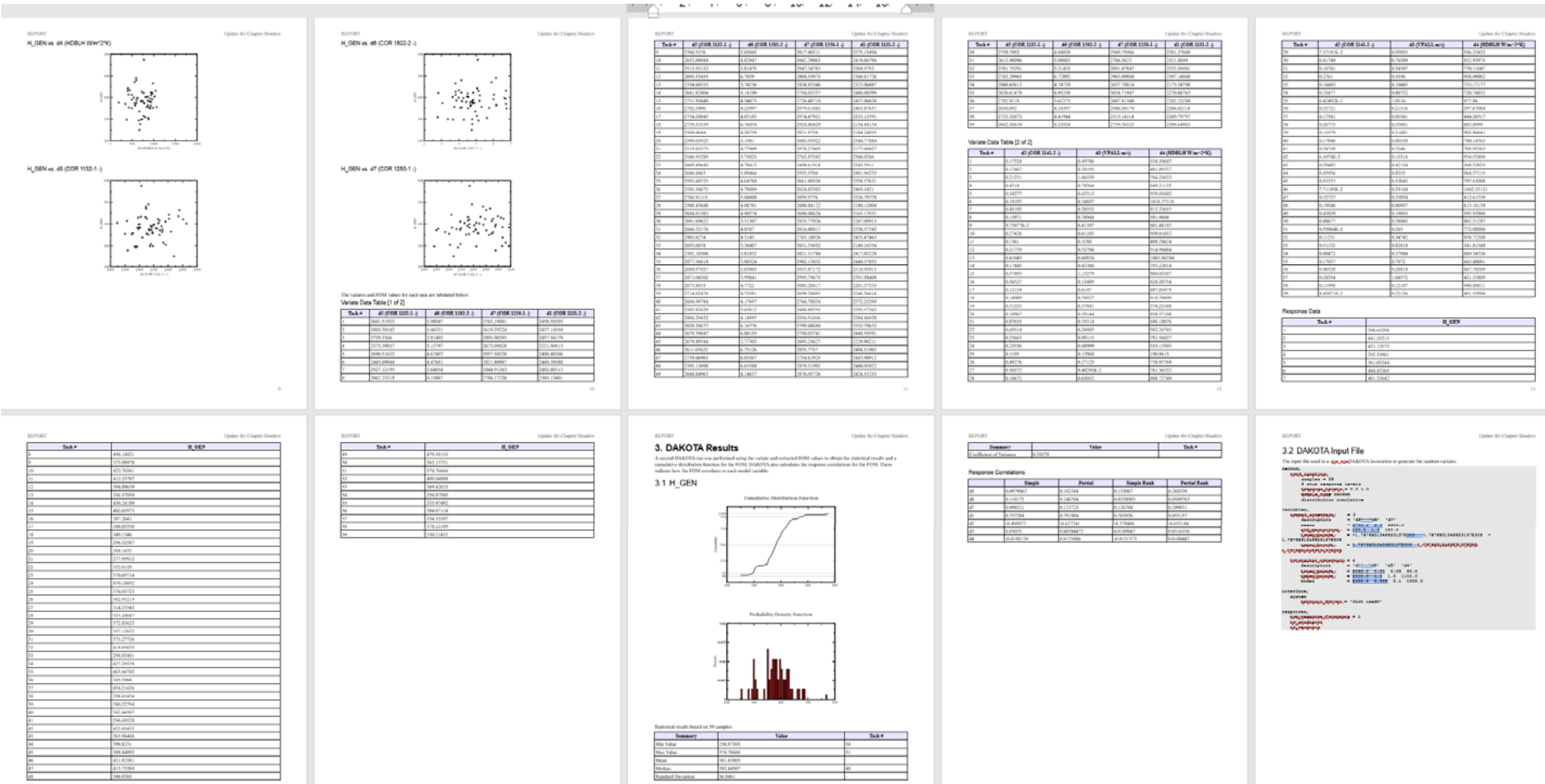
Response Correlations

	Simple	Partial	Simple Rank	Partial Rank
d5	0.0979063	0.102384	0.135067	0.268539
d6	0.110172	0.146704	0.0338983	0.0599765
d7	0.098521	0.123725	0.126768	0.209831
d1	0.737284	0.791904	0.765926	0.855157
d2	-0.495872	-0.627241	-0.379486	-0.653184
d3	0.03035	0.00594472	0.0189947	0.0516358
d4	-0.0188159	0.0125986	-0.0151373	0.0108442

DAKOTA REPORT AUTOMATICALLY GENERATED AT THE END OF THE UNCERTAINTY STREAM APPLICATION



DAKOTA REPORT AUTOMATICALLY GENERATED AT THE END OF THE UNCERTAINTY STREAM APPLICATION



ENEA ACTIVITY IN THE FASTNET – FAST NUCLEAR EMERGENCY TOOL

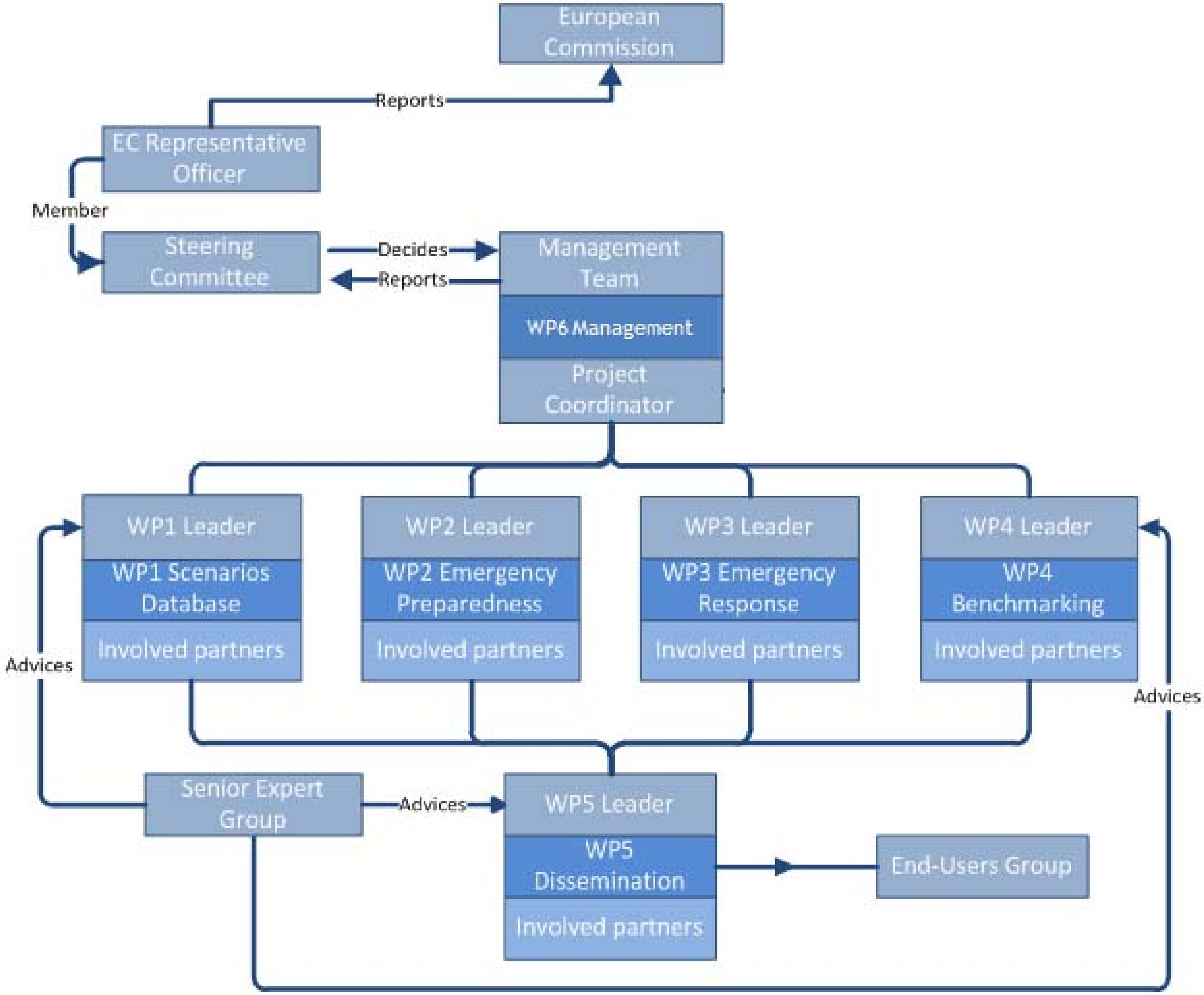
FASTNET – FAST NUCLEAR EMERGENCY TOOL

- FASTNET – FAST NUCLEAR EMERGENCY TOOL – is funded *from the H2020 Framework Programme of the European Commission. The project is coordinate by IRSN;*
- FASTNET Objectives:
  - ✓ To set-up a severe accident scenarios database;
  - ✓ To qualify a common graduated response methodology that integrates several tools and methods to :
    - ✓ evaluate the source term;
    - ✓ ensure both diagnosis and prognosis of severe accident progression;
    - ✓ make the connection between the FASTNET tools and others systems that use source term definition for further assessments in order to implement in any emergency Centres the proposed solution for the management of emergency in all the operating nuclear power plant concepts.
  - ✓ to propose through the project website an action of communication to the public of the emergency management approaches, measures and resources in Europe.

EU-FASTNET PROJECT STRUCTURE

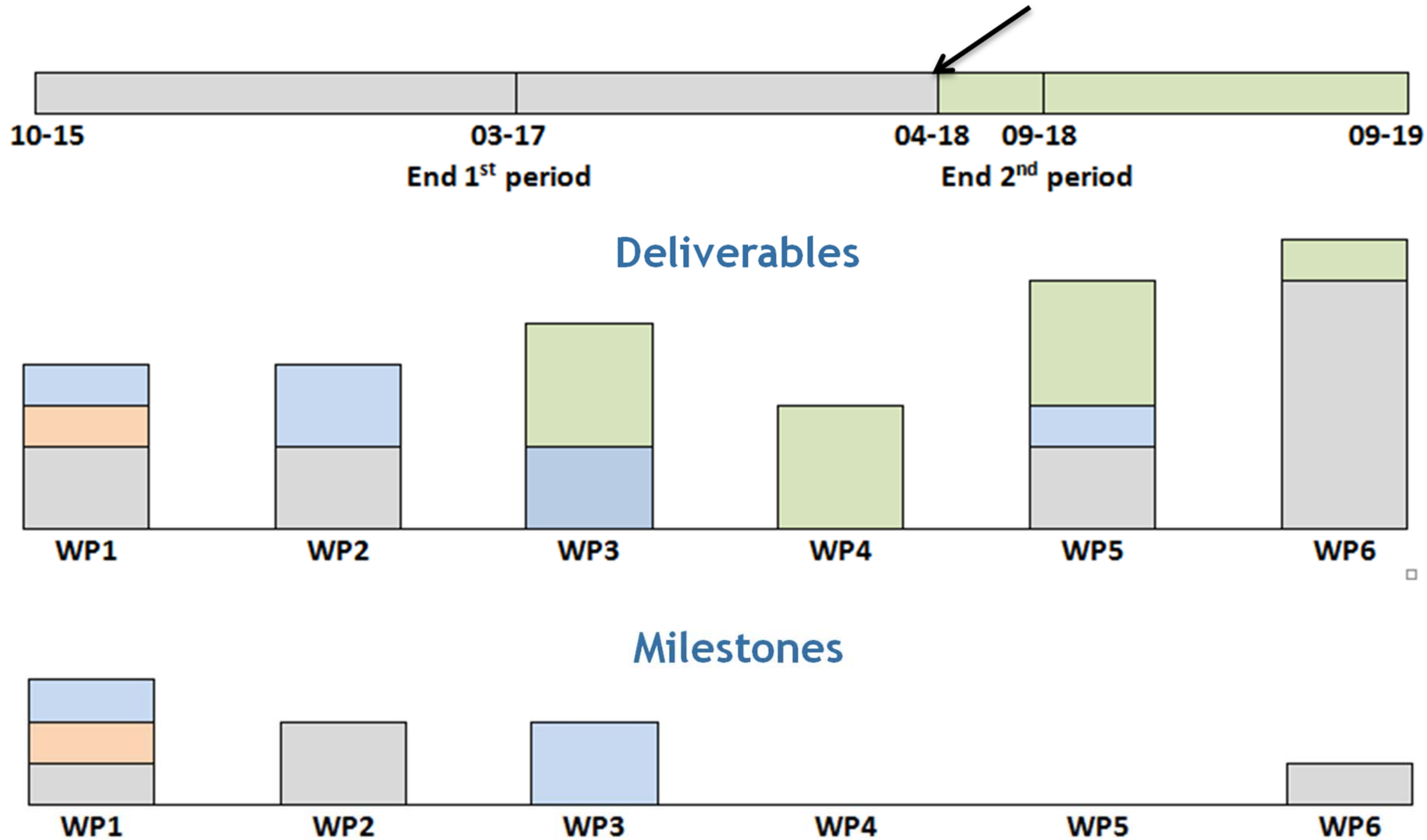
WP	Name/Lead	Description
WP1	**Scenarios database** (LEI)	Elaboration of a common database of pre-calculated scenarios on all concepts of existing NPPs in Europe including the SFP facilities
WP2	**Emergency preparedness** (LRC)	Evaluation and improvement of 2 types of existing approaches: the deterministic approach (3D/3P) and approaches based on BBN
WP3	**Emergency response** (IRSN)	Development of specific parameterisations files describing all concepts of existing NPPs in Europe including SFP facilities which will be included with the PERSAN tool to allow the fast calculation of source terms for any situation Improvement of the BBN approaches to foster their implementation in emergency centres
WP4	**Emergency exercises** (NRPA)	Preparation and the realisation of 2 series of emergency exercises: - the best evaluation of the on-going situation, its evolution and its consequences - the population protection
WP5	**Dissemination** (ENEA)	Sharing of knowledge, including a scenarios database and reference methods and tools beyond the Consortium Education and training through workshops
WP6	**Management** (IRSN)	Project overall administrative and financial management

EU-FASTNET PROJECT STRUCTURE



FASTNET CONSORTIUM

FASTNET OVERVIEW (APRIL 2018)



EUG (April 2018)

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SCENARIOS DATABASE CONCEPT OVERVIEW

- In order to support the development of the selected fast running tools, a suitable reference database has been developed with the State-of-the-Art SA code;

- This database can be used to benchmark and analyze the capability of the fast running codes to predict source terms related to postulated SA sequences;

- Having in mind the need of emergency preparedness, and based on the work done by the FASTNET Senior Expert Group setup in the project, a set of most representative scenarios (LBLOCA, SBO, etc..) has been suggested to the project partners trying to cover wide range of scenarios of generic plant designs found in Europe and other participating countries (e.g. BWR Mark-I like, CANDU like, PWR 900 and 1300 like, VVER 400 and 1000 like, etc);

- Along the project the list developed by the Senior Expert Group has been reviewed by project partners and a final list of representative scenarios has been defined to be included in the database;

- The institutions involved in the database development are ABMERIT, BOKU, CIEMAT, CNSC, ENEA, IRSN, JRC, LRC, NRI, RATEN, SECNRS. The SA code used for the development of the database are ASTEC, MAAP and MAAP-CANDU, MELCOR.

SCENARIOS MATRIX CONSIDERED FOR EACH GENERIC DESIGN

GENERIC DESIGNS	ATW	LBLOCA	IB_LOCA	SB_LOCA	SBO	SGTR	SFP
BWR-MARK1			*		*		
BWR-ABB	*	*			*		
CANDU		*		*	*	*	
French REP 1300		*	*	*			
French PWR-900					*		
PWR-1000		*	*	*	*		*
VVER-440		*			*	*	
VVER- 1000				*	*		

ENEA Contribution by using MELCOR 2.2

INTRODUCTION: SBO TRANSIENT MAIN CHARACTERISTICS

- The ENEA activity related to the severe accident code application in the WP1 of the EU-FASTNET project is focused on the use of MELCOR code to simulate an *"Unmitigated Station Black-Out (SBO)"* to develop a first evaluation of source term released from the containment to the environment.

- The SBO transient is unmitigated and the Start Of the Transient (SOT) is characterized by:
  - Loss of offsite Alternating Current (AC) power:
  - Failure of all the diesel generators;

  Therefore:

  - PRZ level control is unavailable;
  - RCP seal injection is unavailable;
  - Active safey injection systems are unavailable;
  - Motor-driven Auxiliary Feedwater (MDAFW) system is unavailable;
  - Auxiliary feed water is unavailable.

CODE APPLICATIONS: CALCULATION PERFORMED

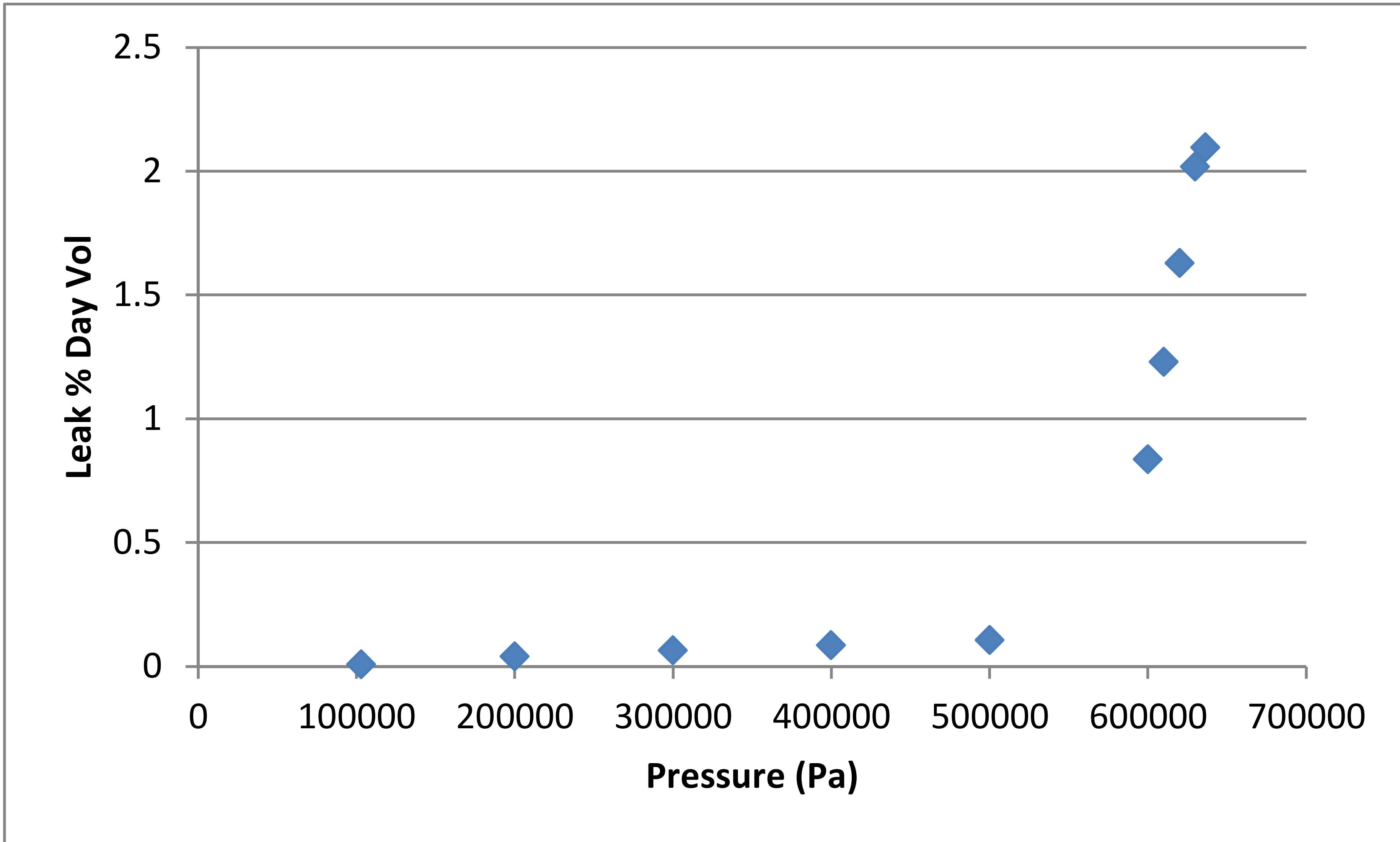
- After the steady state analyses two calculations have been performed:

  - Unmitigated SBO with a containment leak of about 0.1 %Vol day at design pressure (leakage in general increase with the increase of PCV pressure and could preclude the large PCV rupture or failure)
    (REF Label);
  - Unmitigated SBO with a containment leak of about 0.1 %Vol day at design pressure coupled with a postulated containment rupture when the PCV reaches 6 bar.
    (SEN1 label)

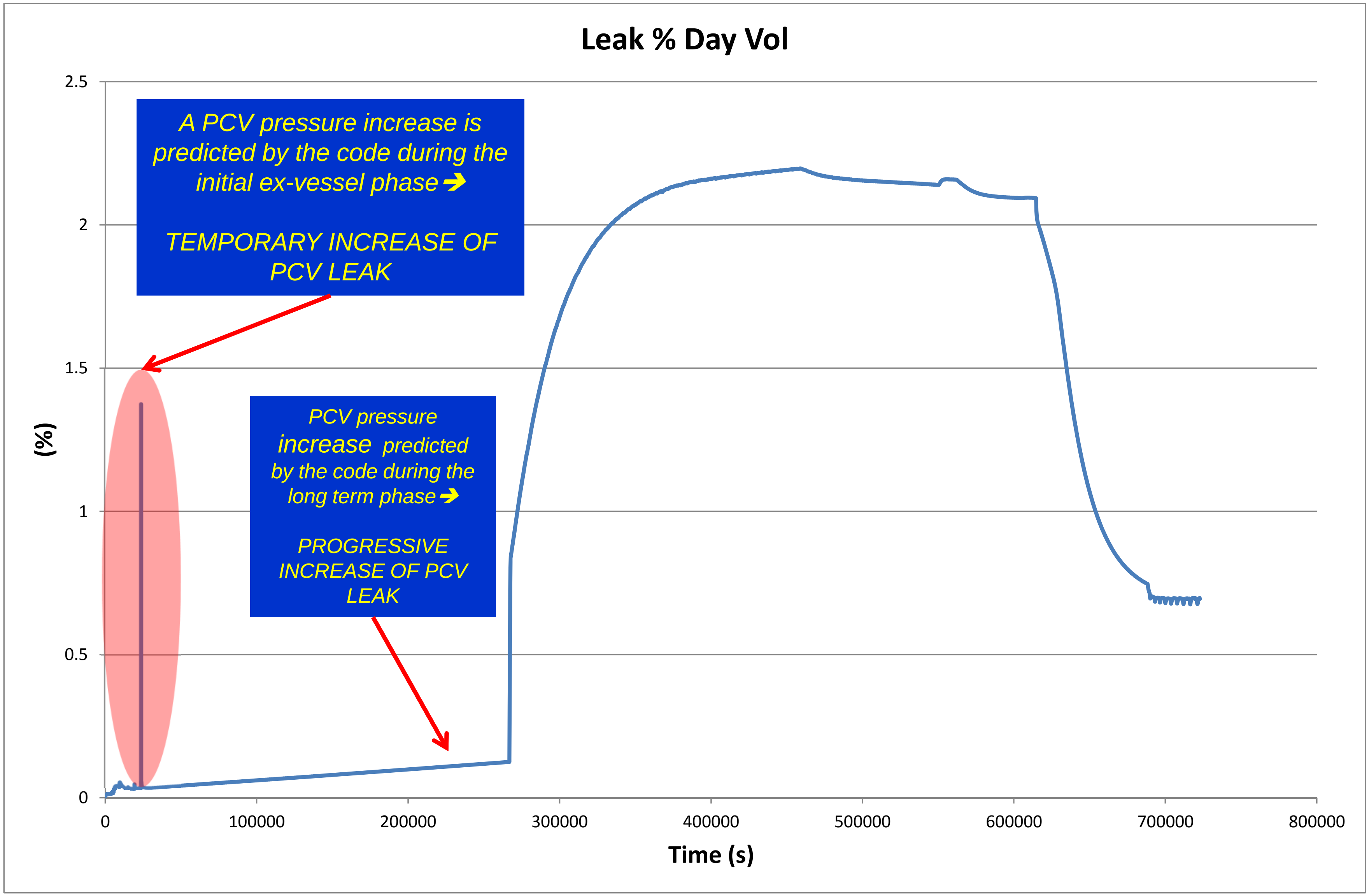
- A generic PCV leakage in % day of PCV volume VS PCV pressure has been implemented by the code user.

- The two calculations are characterized by the same core degradation considering that the main differences take place during the ex-vessel phase due to the postulated containment rupture due to the pressure peak predicted by the code.

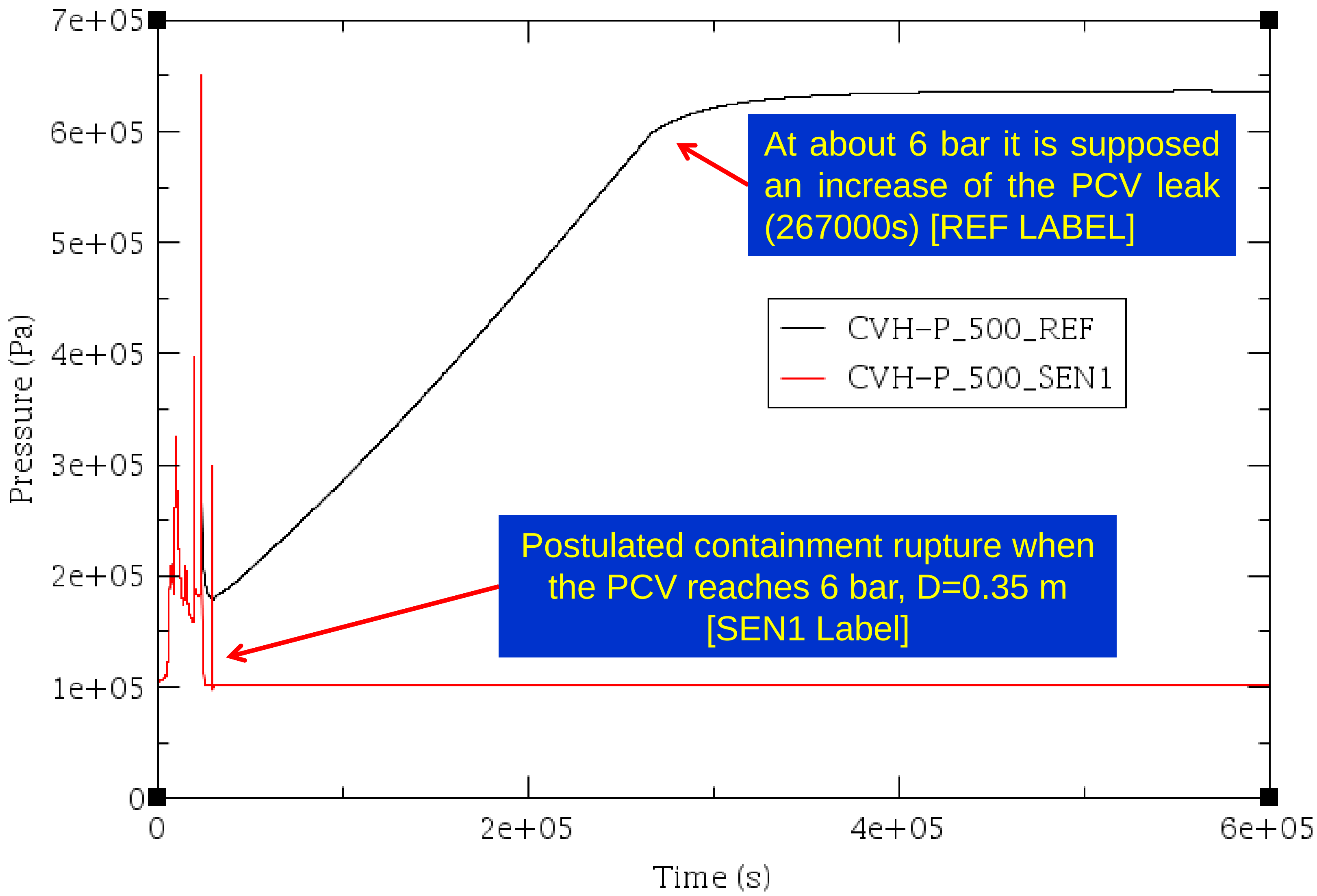
CODE APPLICATIONS: GENERIC PCV LEAKAGE (REF CASE) IN % DAY OF PCV VOLUME VS PCV PRESSURE



CODE APPLICATIONS: GENERIC PCV LEAKAGE (REF CASE) IN % DAY OF PCV VOLUME VS TIME



CODE APPLICATIONS: PCV PRESSURE BEHAVIOR FOR THE REF AND SEN1 CALCULATION



SIMPLE USER APPROACH TO CALCULATE ACTIVITY RELEASED FROM MELCOR CODE IN A TARGET VOLUME

- This is a simple and simplified approach used to convert RN mass class, released to a target control volume, to activity;
- The first step is to calculated the initial mass of *the isotope i* at the shutdown of the reactor (Mi);
- Then it is calculated the percentual composition (%) referring to the correspondent RN chemical class x mass (Mx) at the shutdow (Mi/Mx);
- Then the mass of MELCOR RN class x released in the target volume is directly multiplied for this fraction

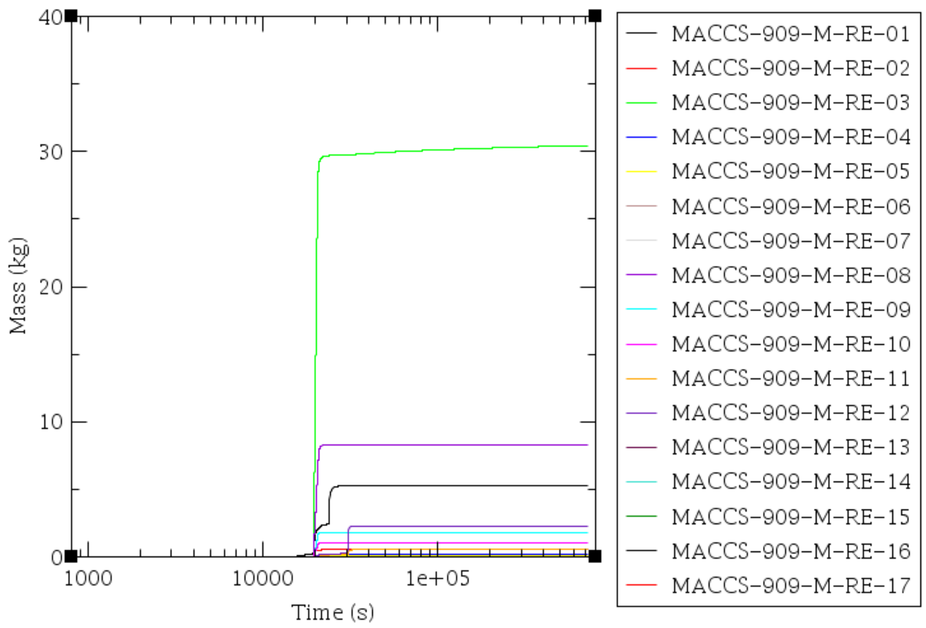
  - ✓ *mi is the isotopic mass released in the selected volume*
  - ✓ *$M_{RN,x}$ is the mass of MELCOR RN class x released in the selected volume (environment in this case)*
  - ✓ *Mi is the initial core inventory at the shutdown of the isotope i*
  - ✓ *Mx is the initial inventory at the shutdown of the class x*

$$mi = M_{RN,x}\frac{M_i}{M_x}$$

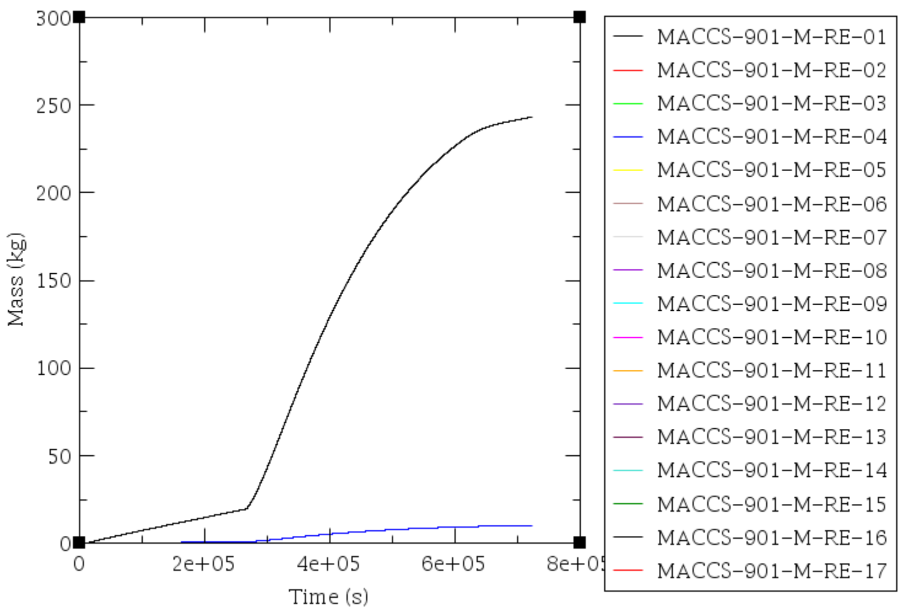
- In general the initial mass of *the isotope i* at the shutdown is calculated by using Origen; but in this case has been postulated as the same percentual composition of the Surry Plant presented in the NUREG/CR-7110. This a strong approximation but the target of this analyses is only to give an indicative value of the activity released.

CODE APPLICATION: RELEASE IN RN CLASS (REF CASE)

Source term from the cavity

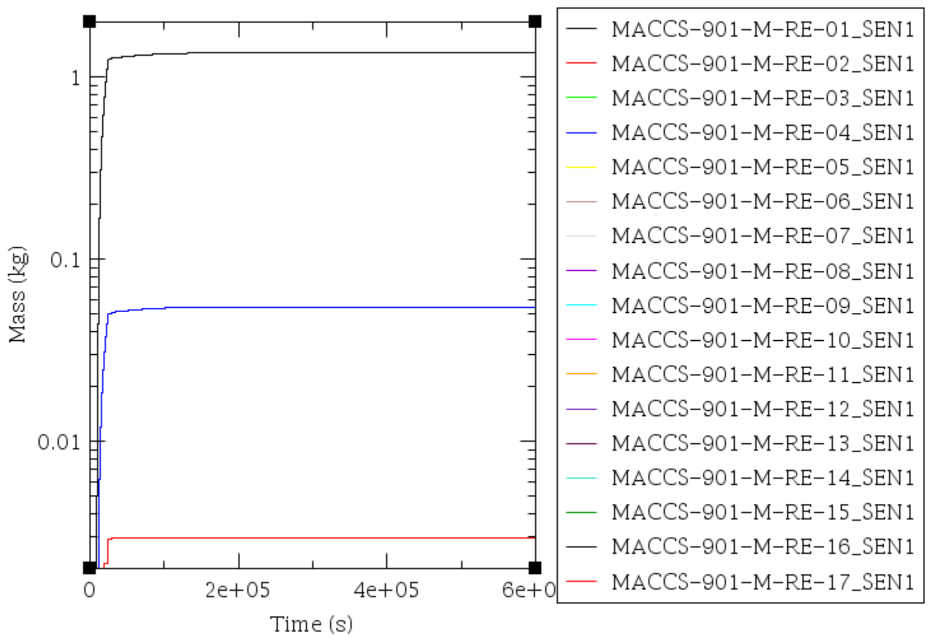


Source term to the environment from the PCV leakage

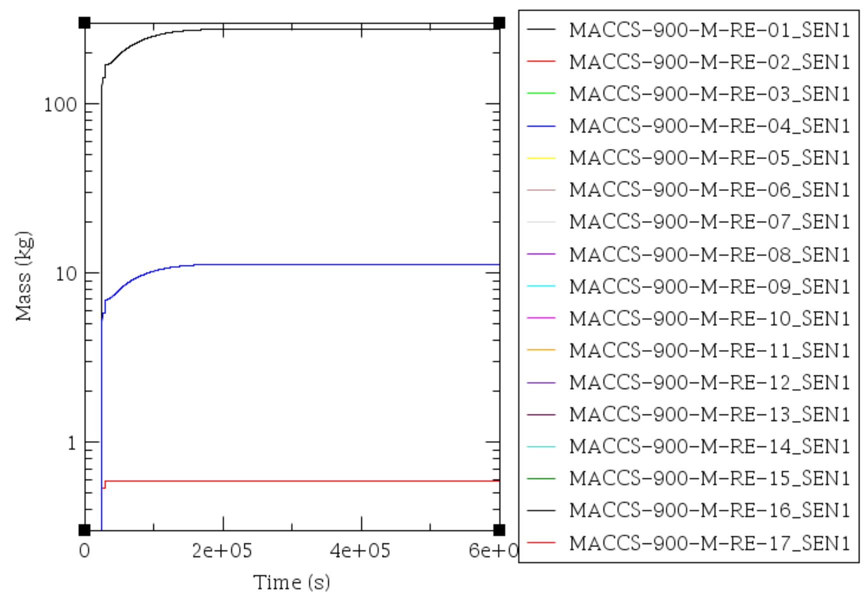


CODE APPLICATION: RELEASE IN RN CLASS (SEN1 CASE)

Source term from the leakage



Source term from the rupture



CODE APPLICATION: RELEASE IN RN CLASS

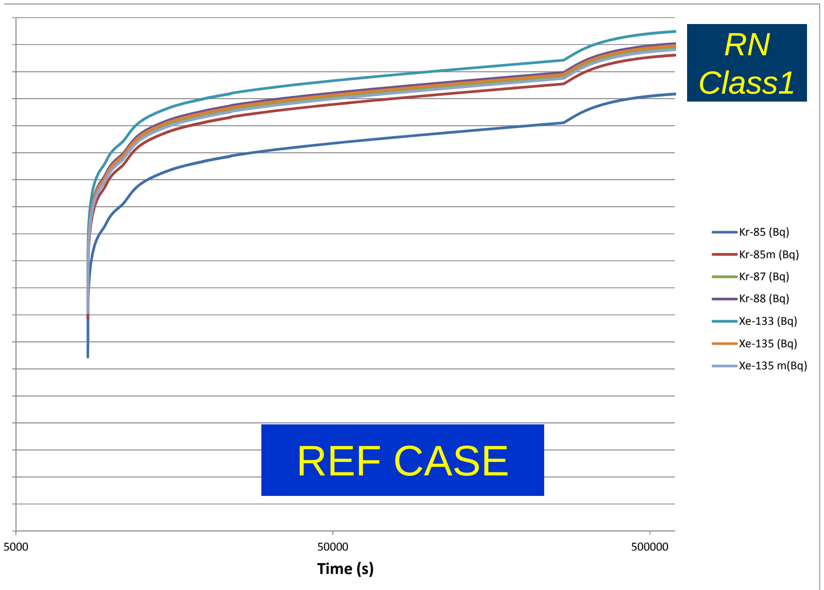
REF

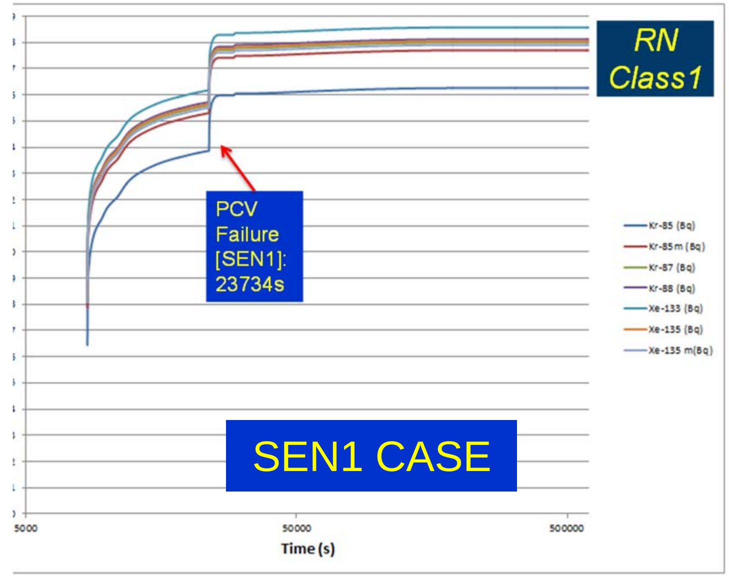
CLASS NAME	Representative	INPUT MELCOR (kg) (shutdown)	Released (kg)	Released (%)
NOBLE GAS	Xe	278.3	2.4324E+02	8.7402E-01
ALKALI METALS	Cs	155.1	3.0436E-03	1.9624E-05
ALKALINE EARTH	Ba	122.1	5.5711E-04	4.5628E-06
HALOGENS	I	11.99	9.9340E+00	8.2852E-01
CHALCOGENS	Te	24.42	2.6705E-04	1.0936E-05
PLATINOIDS	Ru	171.8	9.9786E-15	5.8082E-17
EARLY TRANSITION ELEMENTS	Mo	202.6	1.8662E-13	9.2111E-16
TETRAVALENT	Ce	357.4	5.2334E-07	1.4643E-09
TRIVALENTS	La	331.6	3.5440E-07	1.0688E-09

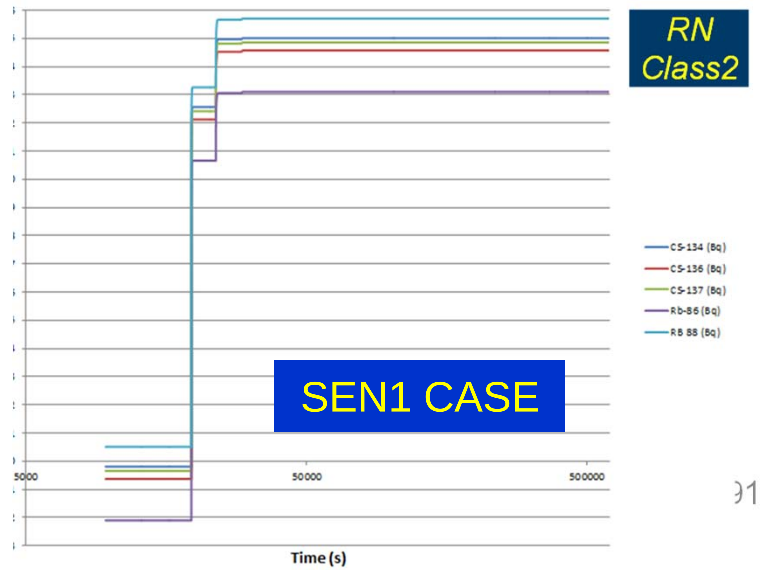
SEN1

CLASS NAME	Representative	INPUT MELCOR (kg) (Stutdown)	Released (kg)	Released (%)
NOBLE GAS	Xe	278.3	2.7832E+02	1.0001E+00
ALKALI METALS	Cs	155.1	5.9626E-01	3.8444E-03
ALKALINE EARTH	Ba	122.1	2.2217E-01	1.8196E-03
HALOGENS	I	11.99	1.1367E+01	9.4802E-01
CHALCOGENS	Te	24.42	6.9930E-02	2.8637E-03
PLATINOIDS	Ru	171.8	6.3752E-13	3.7108E-15
EARLY TRANSITION ELEMENTS	Mo	202.6	1.1762E-11	5.8053E-14
TETRAVALENT	Ce	357.4	3.7091E-05	1.0378E-07
TRIVALENTS	La	331.6	2.5068E-05	7.5598E-08

RELEASE TO THE ENVIRONMENT RADIONUCLIDE INVENTORY AT THE END OF CALCULATION







CONCLUSIONS

- The ENEA activity related to the severe accident code application in the WP1 pf the EU-FASTNET project is focused on the use of MELCOR code to simulate two *"Unmitigated Station Black-Out (SBO)"* calculations to develop a first evaluation of source term released from the containment to the environment

- MELCOR PWR-900 like nodalization, developed by using SNAP, was designed to have a reasonable computational time and a realistic prediction of the thermal hydraulic and degradation phenomena involved during a postulated transient assuring a reliable and accurate transient simulation.

- The input deck has been tested during the EU-CESAM project by doing an unmitigated SBO transient and comparing the results with ASTEC and MAAP calculations in relation to the in-vessel phase. The results of the calculated data show that the three codes predict the phenomenological evolution in a good qualitative agreement though with some quantitative differences mostly related to the in-vessel Hydrogen generation.

CONCLUSIONS

- The development of the nodalization is still in progress and has been continued during this EU-FASTNET project reaching a revision 2 of the nodalization. In particular a complete review of the CAVITY and RADIONUCLIDE Package nodalization has been done in order to give a first estimation of the potential source term release to the environment in a postulated unmitigated transient. This is the first source term estimation done with this nodalization.

- The Source term has been successfully calculated by MELCOR code in RN Class.

- A simple and simplified approach used to convert RN mass class, released to a target control volume, to activity has been used.

- The resuts of the calculated, as expected, show that an early failure of the containment determine a major release of radionuclide to the environment.

- The develpoment of the MELCOR nodalization is still in progress and more detail nodalization is planned, for future research activities, to be used to model separately the different compartment of the PWR containment.

CONCLUSIONS

- The results of the activity have been presented in:

  - F. Mascari, Source term evaluation with melcor code in the eu-fastnet project framework, The 10th Meeting of the “European MELCOR User Group” Faculty of Electrical Engineering and Computing (FER), University of Zagreb, Unska 3 Zagreb, Croatia, 25th-27th April, 2018.

  - F. Mascari, F. Rocchi P. Carny, L. Liptak, M. Adorn, J. Fontanet, L.E. Herranz, M. Shawkat, W. Raskob, F. Cousin, J. C. de la Rosa Blul, E. Urbonavicius, F. Di Dedda, T. Augustsson, M. Constantin, G. Arbaev, P. Isaksson, J. Kubicek, FASTNET SCENARIOS DATABASE DEVELOPMENT AND STATUS, The 9th European Review Meeting on Severe Accident Research (ERMSAR 2019), Clarion Congress Hotel, Prague, Czech Republic, March 18-20, 2019

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ENEA AND JRC JOINT ACTIVITY IN THE WP40-SAM

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- In the framework of the European Project CESAM - Code for European Severe Accident Management- coordinated by GRS, ENEA is using MELCOR 2.1 code in the **WP40** - Plant applications and Severe Accident Management (**SAM**), coordinated by JRC. In particular, ENEA is involved in the development of a "PWR 900 like" with MELCOR code for benchmarking the ASTEC code.

- Within this CESAM framework, ENEA and JRC have started a joint research activity focused on the analysis of an unmitigated station blackout with MELCOR (analyses developed by ENEA) and MAAP (analyses developed by JRC) code in order to benchmark ASTEC code (analyses developed by JRC).

- Activity is performed in collaboration with JRC. Contacts:
  - Fulvio Mascari - ENEA
  - Giacomino Bandini - ENEA
  - Marco Sangiorgi - JRC
  - Juan-Carlos DE-LA-ROSA-BLUL - JRC

ENEA AND JRC JOINT ACTIVITY IN THE WP40-SAM: UNMITIGATED SBO

- The activity is focused on the use of MAAP and MELCOR code to simulate an *"Unmitigated Station Black-Out (SBO)"* to benchmark ASTEC code.

- The SBO transient is unmitigated and the Start Of the Transient (SOT) is characterized by:
  - Loss of offsite Alternating Current (AC) power:
  - Failure of all the diesel generators;

Therefore:

- PRZ level control is unavailable;
- RCP seal injection is unavailable;
- Active safey injection systems are unavailable;
- Motor-driven Auxiliary Feedwater (MDAFW) system is unavailable;
- Auxiliary feed water is unavailable.

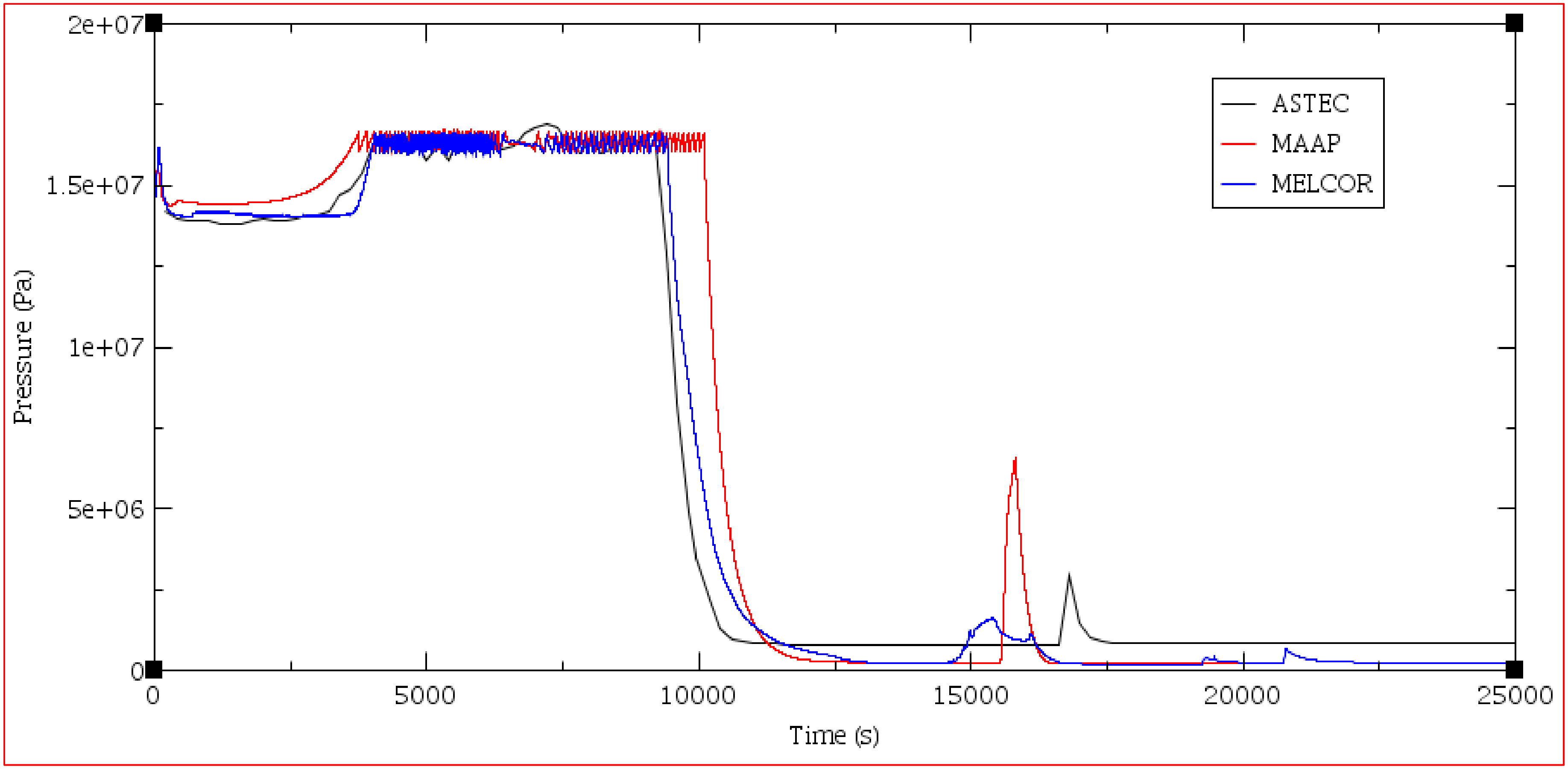
CODE CALCULATIONS:ASTEC, MAAP AND MELCOR CALCULATED DATA COMPARISON

RELEVANT PHENOMENOLOGYCAL ASPECTS	ASTEC	MAAP	MELCOR	MAAP DISCR* (%)	MELCOR DISCR* (%)
SG1,2,3 Cycling Inception (s)	200	100	30	-	-
SEBIM Cycling Inception (s)	4200	3757	4058	10.55	3.38
Two Phase inception in the HL (s)	6400	6404	6300	0.06	1.56
Core TAF Uncovered (s)	8000	8083	7000	1.04	12.50
H2 Start (s)	8400	8795	8382	4.70	0.21
SEBIM Stuck Open (s)	9200	10099	9414	9.77	2.33
Core BAF Uncovery (s)	9400	10165	9570	8.14	1.81
TCL 1300K (s)	9970	10845	8700	8.78	12.74
TCL 1855K (s)	10080	10904	9248	8.17	8.25
Upper Core Ring Failure 1 (s)**	10953	12786	11600	16.74	5.91
Upper Core Ring Failure 2 (s)**	10953	12724	13100	16.17	19.60
Upper Core Ring Failure 3 (s)**	11353	12866	13380	13.33	17.85
Upper Core Ring Failure 4 (s)**	11753	13484	13650	14.73	16.14
Upper Core Ring Failure 5 (s)**/***	12353	14815	14380	19.93	16.41
Slumping Inception (s)	16600	15526	14580	6.47	12.17
Vessel Failure (s)	18157	20608	19250	13.50	6.02

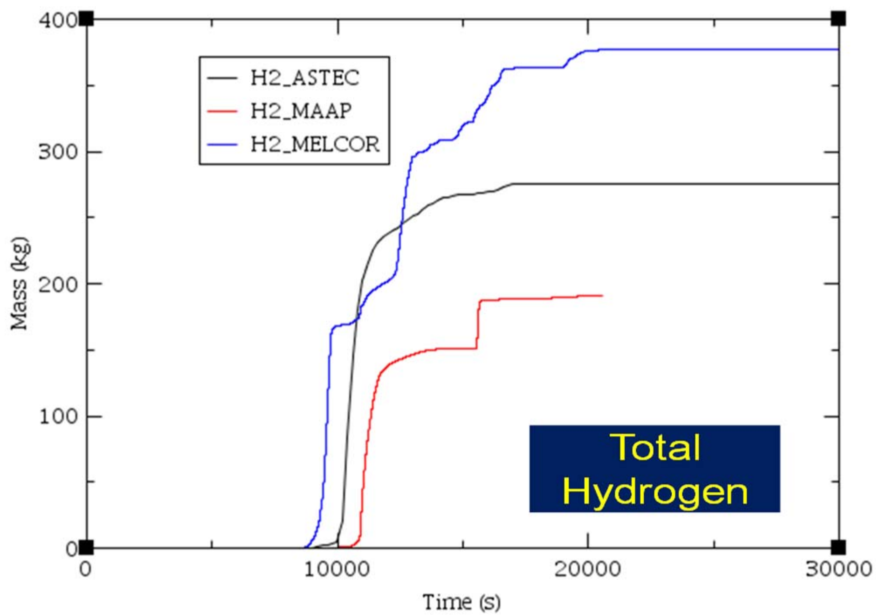
**ASTEC calculated data discrepancies based on the comparison with MAAP and MELCOR calculated data.*
***For ASTEC it is estimate the instant when the fuel ring continuity is lost.*
*** For MELCOR calculation, the upper part of the 5th ring starts to collapse at 14380s, but other axial levels continue their failure starting from 15270s.*

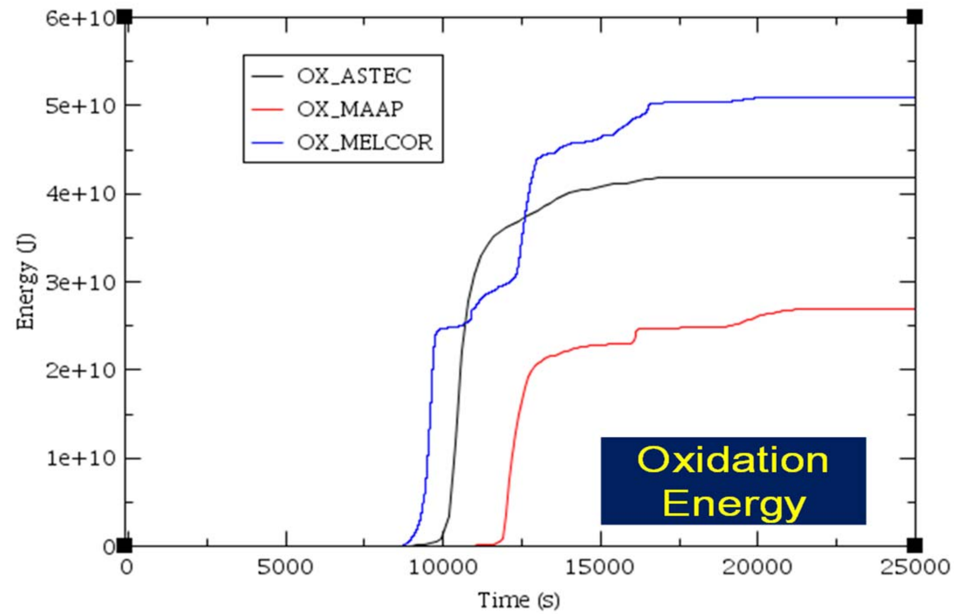
ENEA

CODE CALCULATIONS:ASTEC, MAAP AND MELCOR CALCULATED DATA COMPARISON

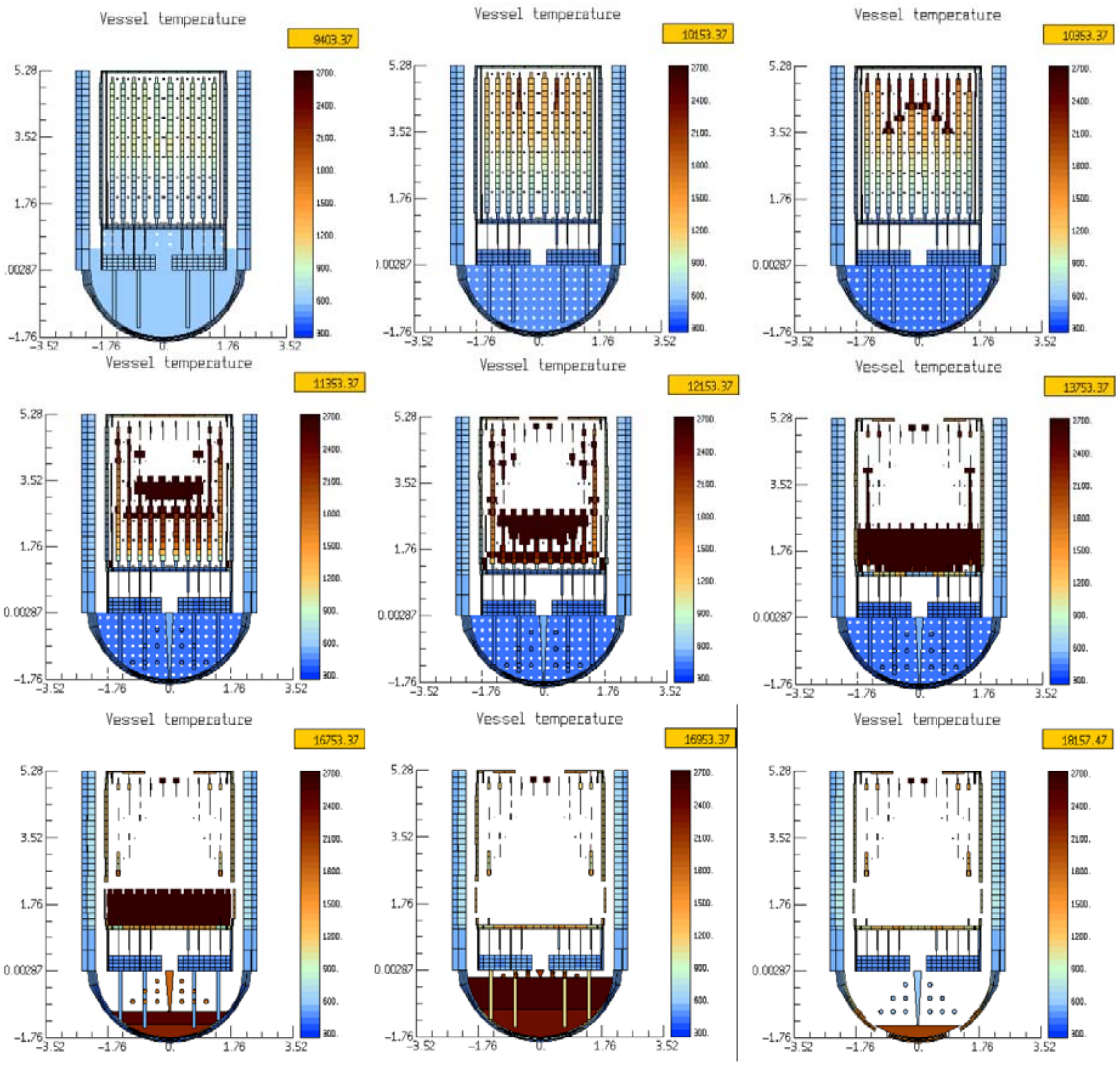


CODE CALCULATIONS:ASTEC, MAAP AND MELCOR CALCULATED DATA COMPARISON



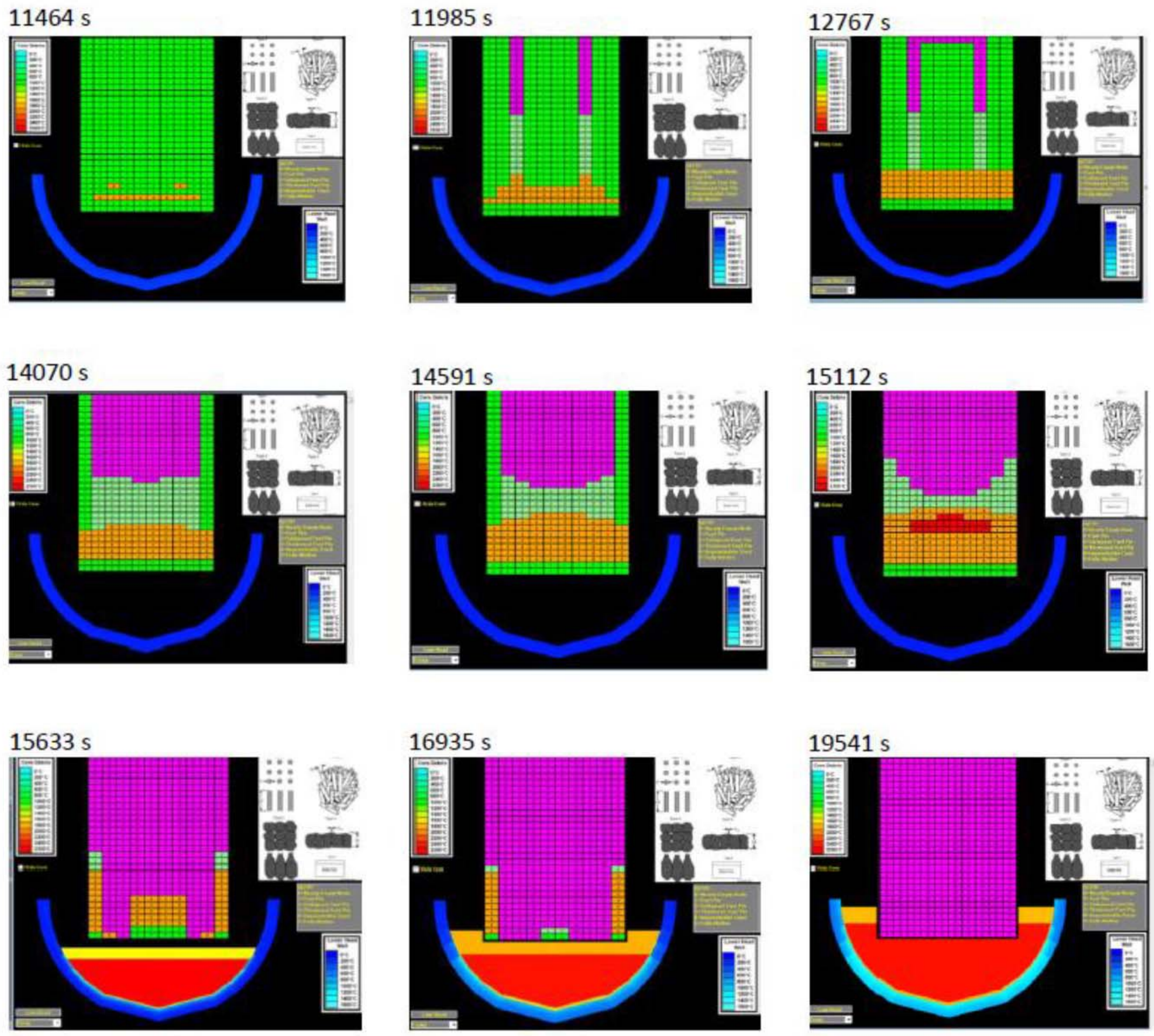


ASTEC CORE DEGRADATION REPRESENTATION



ASTEC 2.1

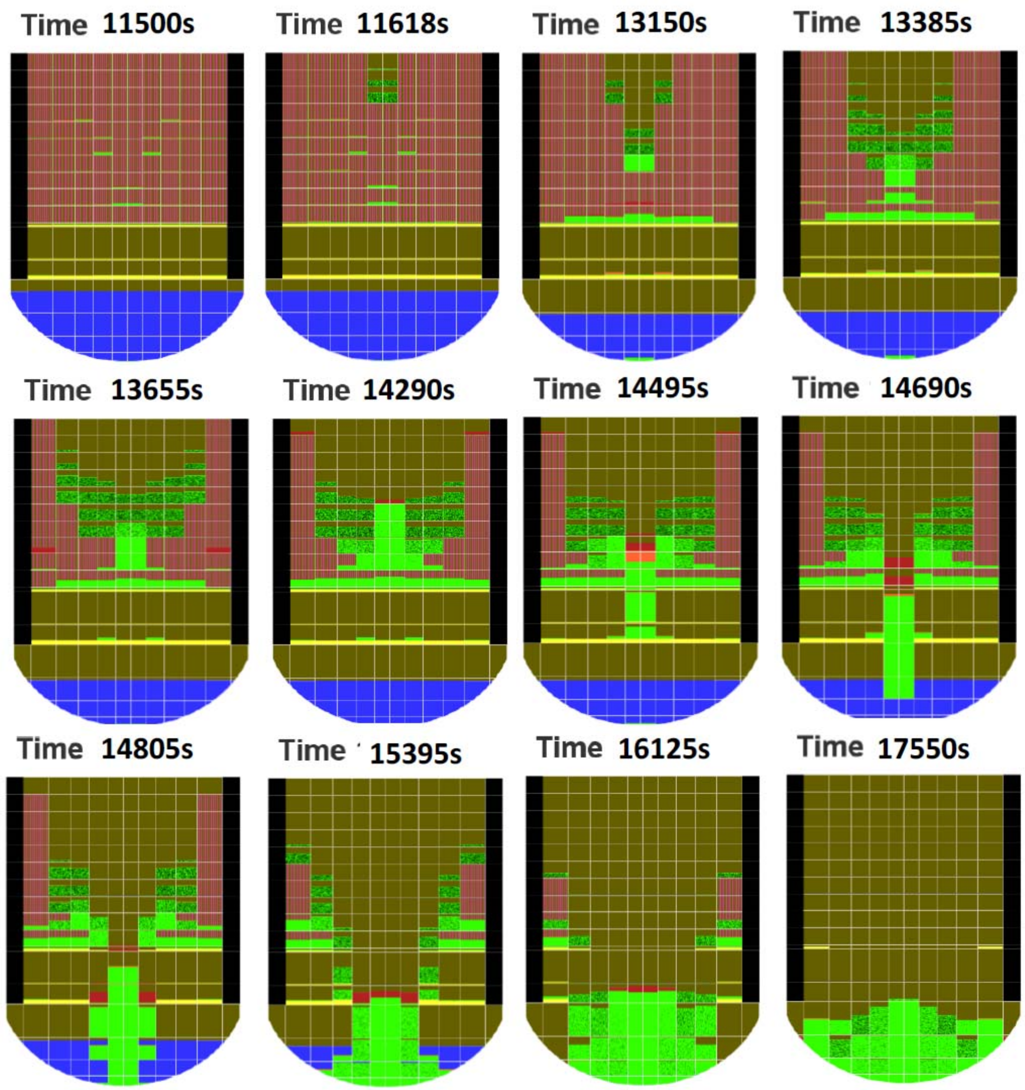
MAAP CORE DEGRADATION REPRESENTATION



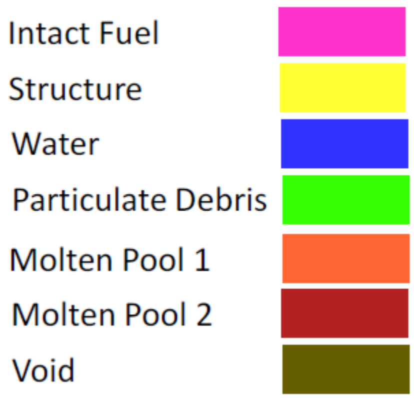
MAAP 5.02

MAAP core degradation representation (number 0,1,2,3,4,5 represent the type of degradation that take place in each node; in particular : 0= Nearly Empty Node; 1=Fuel Pin; 2= Collapsed Fuel Pin; 3=Thickened Fuel Pin; 4= Impenetrable Crust; 5= Fully Molten).

MELCOR CODE DEGRADATION PHASES REPRESENTATION MADE BY USING SNAP



MELCOR 2.1



CONCLUSIONS

- The results of the calculated data show that the three codes predict the phenomenological evolution in a good qualitative agreement though with some quantitative differences.

- In particular, considering the time sequence of relevant phenomenological aspects, the maximum percentage discrepancy between ASTEC and MAAP/MELCOR calculated data **is at maximum of about the 20%** for the main selected safety related parameters chosen as figure of merit.

- The most relevant differences are observed in the in-vessel hydrogen mass production prediction. Such discrepancies underline some modeling differences between the three codes related to core material degradation/relocation, determining differences in the available area for the oxidation process, different flow blockage conditions, different code node porosity prediction, etc.

- In addition it is to note a phenomenological discrepancy related to the slumping predictions between ASTEC and MAAP/MELCOR calculations: while MAAP and MELCOR predict a core lower plate failure with a consequent relocation of degraded core material in the lower plenum, ASTEC predicts the relocation of the degraded core material through the shroud failure.

CONCLUSIONS

- Considering
  - the hypotheses of the transient (no ECCS intervention, scram at zero, no pump leakage, etc) and
  - the maximum degree of freedom left to the *Code-User* (hydraulic and core nodalization strategy and degree of detail, setting of the boundary condition…) and
  - the general phenomenological agreement of the transient phenomenology predicted by the three codes (with the exception of the slumping phenomenology)

  the results of the code calculations can be used as a confirmation of the transient phenomenological evolution of the postulated accident.
- Future activity based on a strictly congruence analysis between core structures nodalizations (geometry and mass) is endorsed.

CONCLUSIONS

- *The results of the activity has been presented at "Technical Meeting on the Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors" with a paper titled: F. Mascari, J. C. De La Rosa Blul, M. Sangiorgi, G. Bandini, ASTEC, MAAP AND MELCOR BENCHMARK CODE ANALYSIS OF AN UNMITIGATED SBO TRANSIENT IN A PWR-900 LIKE REACTOR*

- *A NUREG-IA, 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 cases, NUREG/IA-09xx, has been submitted for the publication*

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