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

SEVERE ACCIDENT AND UNCERTAINTY ESTIMATION, NEEDS AND CURRENT ACTIVITIES

MODELLING IN NUCLEAR SCIENCE AND ENGINEERING SEMINAR 2020, 4/5 November 2020 online seminar

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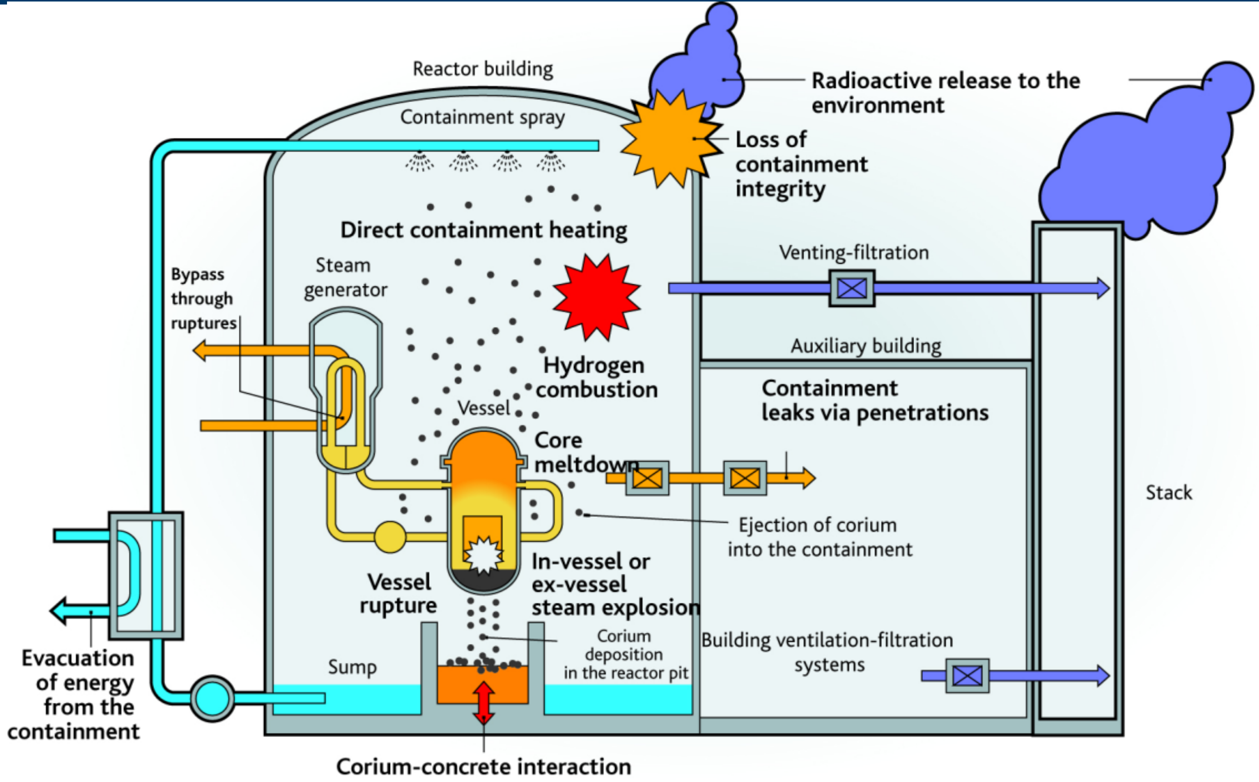
INTRODUCTION

- After the Fukushima accident, the interest of each country using nuclear energy as a part of its national energy mix has been more focused on severe accident mitigation strategies.
- Several severe accident management analyses have been performed to analyze
  - Accident progression;
  - Core damage;
  - Grace period;
  - Fission product release.

  demonstrating the accident management strategy adequacy.
- Nuclear reactors are designed to maintain the fuel damage and radioactive release within authorized limits during selected postulated accident - Design Basis Accident (DBA) -.
- A Severe Accident (SA) is a Beyond Design Basis Accident (BDBA) involving significant core degradation.

MAIN PHYSICAL PHENOMENA DURING A SEVERE ACCIDENT



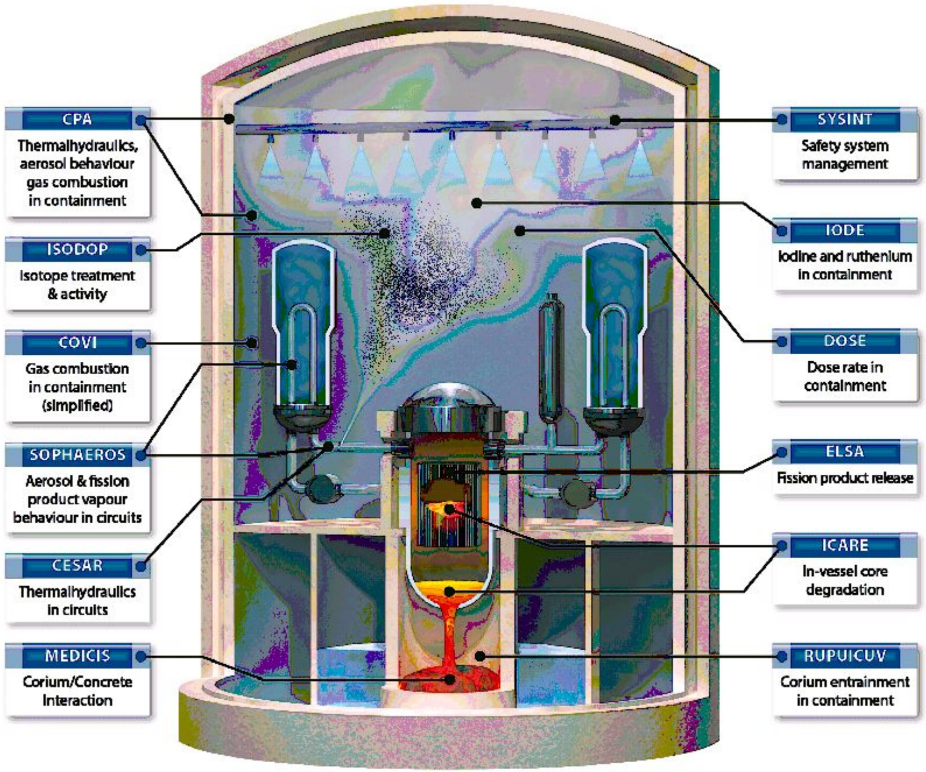
Jean-Pierre Van Dorsselaere, Thierry Albiol and Jean-Claude Micaelli (September 6th 2011). Research on Severe Accidents in Nuclear Power Plants, Nuclear Power - Operation, Safety and Environment, Pavel Tsvetkov, IntechOpen, DOI: 10.5772/17552. Available from: https://www.intechopen.com/books/nuclear-power-operation-safety-and-environment/research-on-severe-accidents-in-nuclear-power-plants

INTRODUCTION

- Several computational tools can be used to analyze a severe accident;
- Considering the complexity and mutual different interacting and interrelated phenomena/processes along a severe accident transient progression, a key role is played by the state-of-the-art severe accident integral codes such:
  - ASTEC;
  - MAAP;
  - MELCOR;
  - etc

  that can be considered the key tool to design a suitable accident management strategy.
- These codes, storing all the knowledge developed in the last decades from the experimental activities, allow confirmation of the transient progression of the modeled plant, during a postulated severe accident, characterizing the main severe accident phenomena taking place in the reactor pressure vessel, reactor cavity, containment, and the confinement buildings typical of LWRs.

EXAMPLE: STRUCTURE OF THE ASTEC INTEGRAL CODE FOR SA SIMULATION



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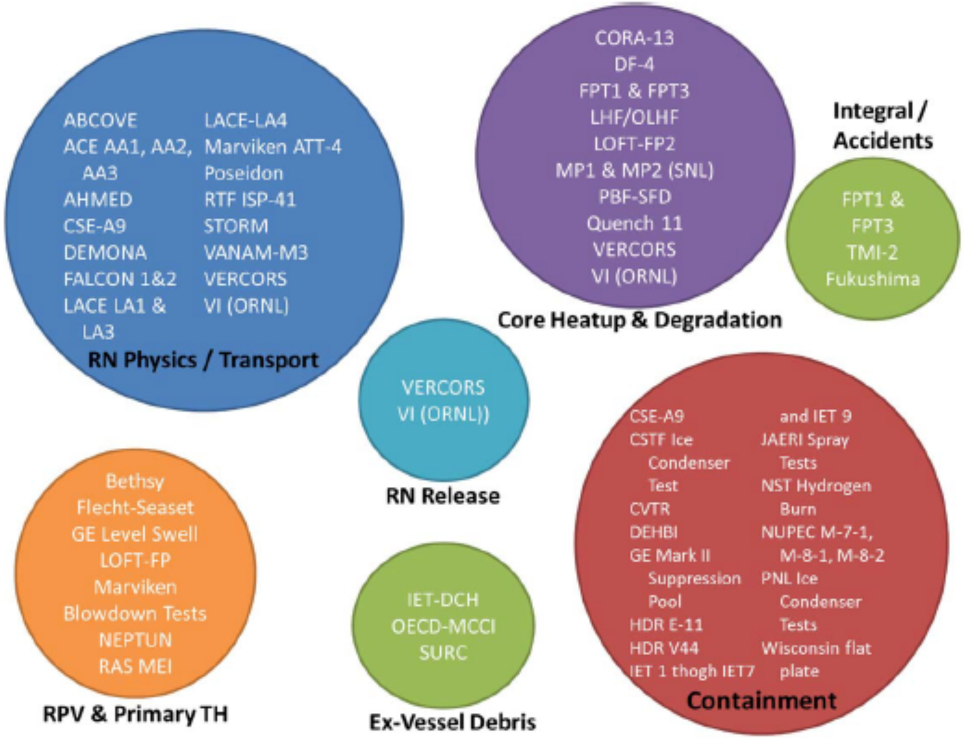
INTRODUCTION

- Several models/correlations have been implemented in these state-of-the-art severe accident codes and must be set by the code user during input deck development.
- <u>Several experimental programs in the field of severe accident phenomena</u> have been conducted to validate the models and correlations implemented in the codes and provided a valuable "<u>assessment database</u>" to assess severe accident simulation tools.

EXAMPLE: EXPERIMENTS/ACCIDENTS USED FOR VALIDATION OF MELCOR



MELCOR Computer Code Manuals Vol. 3: MELCOR Assessment Problems Version 2.1.7347 2015, SAND2015-6693 R, https://www.nrc.gov/docs/ML1530/ML15300A476.pdf

SEVERE ACCIDENT AND UNCERTAINTY ESTIMATION, NEEDS

- The analyses of the current state-of-the-art shows that:
  - There is a need to reduce some uncertainties still present and
  - A consequent investigation of phenomena/processes, to date not investigated in detail in geometric prototypical experimental facility with prototypical material, should be addressed.
- When the validation data exist there is a general agreement between the different codes;
- However discrepancies in some core degradation phenomena can be still observed when comparing the results as predicted by different simulation tools considering the different core degradation models implemented in the codes.

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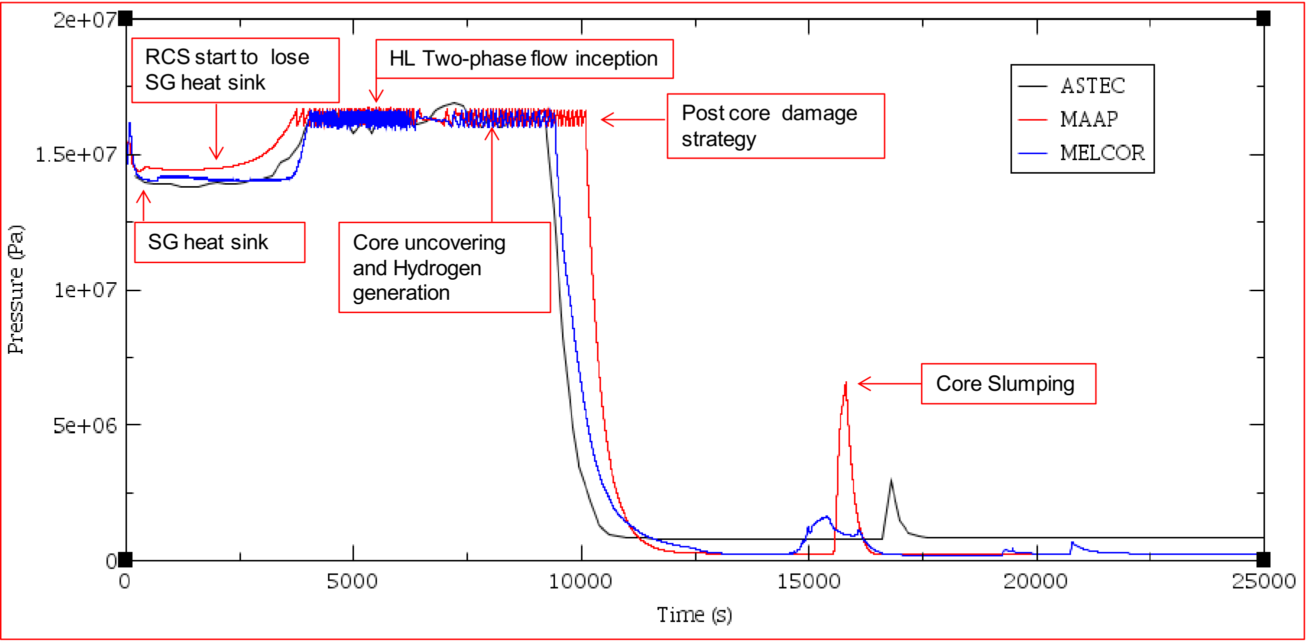
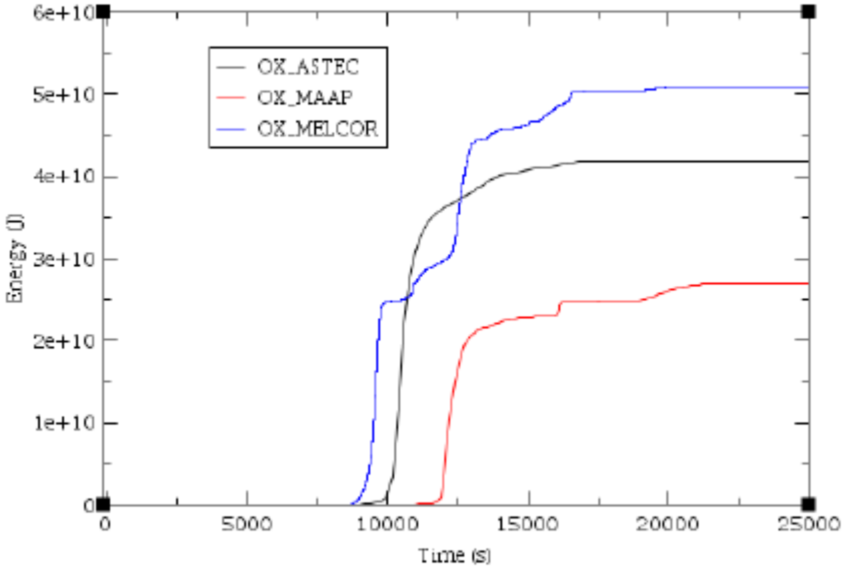
EXAMPLE: Analyses of an Unmitigated Station Blackout Transient in a Generic PWR-900 with ASTEC, MAAP and MELCOR Codes, NUREG/IA-0515



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

EXAMPLE: Analyses of an Unmitigated Station Blackout Transient in a Generic PWR-900 with ASTEC, MAAP and MELCOR Codes, NUREG/IA-0515



- 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.
- It is important to note that the area available for the oxidation has a great uncertainty due to the complex phenomena taking place during the degradation and relocation of the core material and the limited full-scale experimental data for validation purpose.

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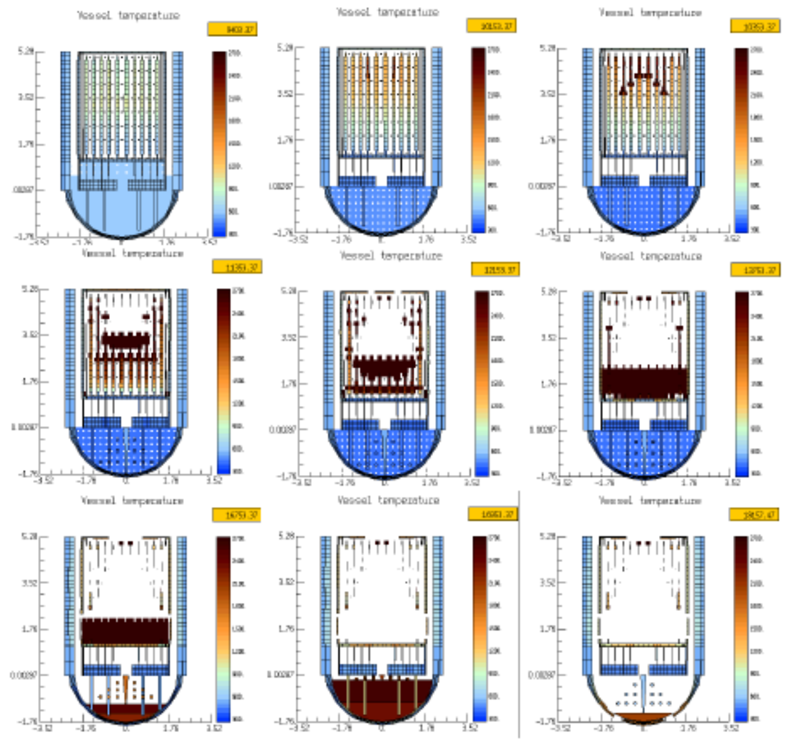
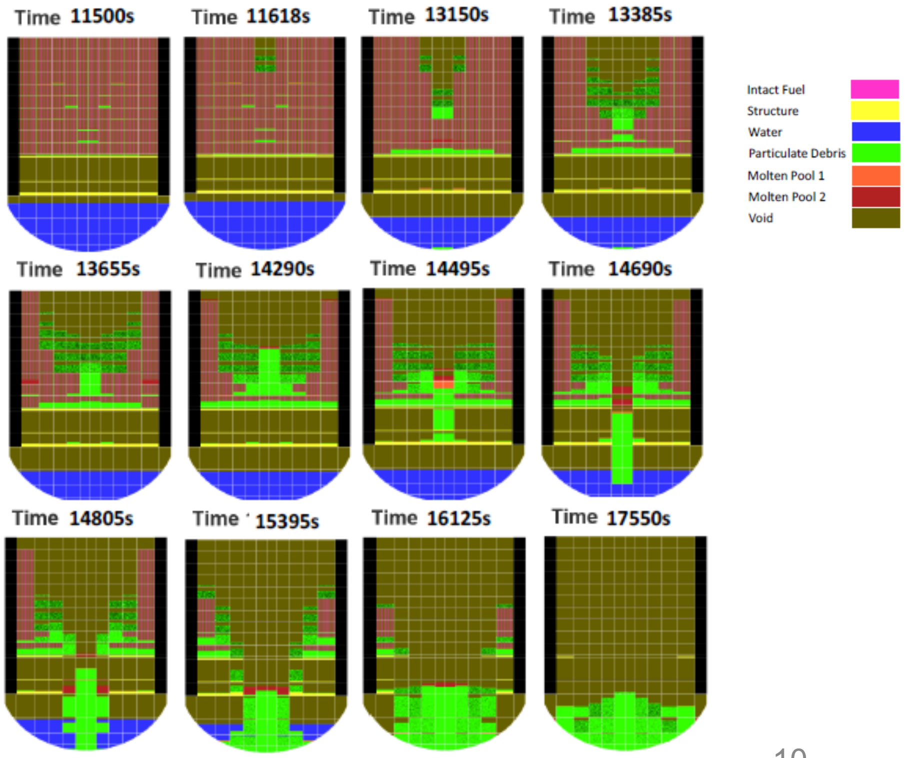
EXAMPLE: Analyses of an Unmitigated Station Blackout Transient in a Generic PWR-900 with ASTEC, MAAP and MELCOR Codes, NUREG/IA-0515

ASTEC Core Degradation Representation

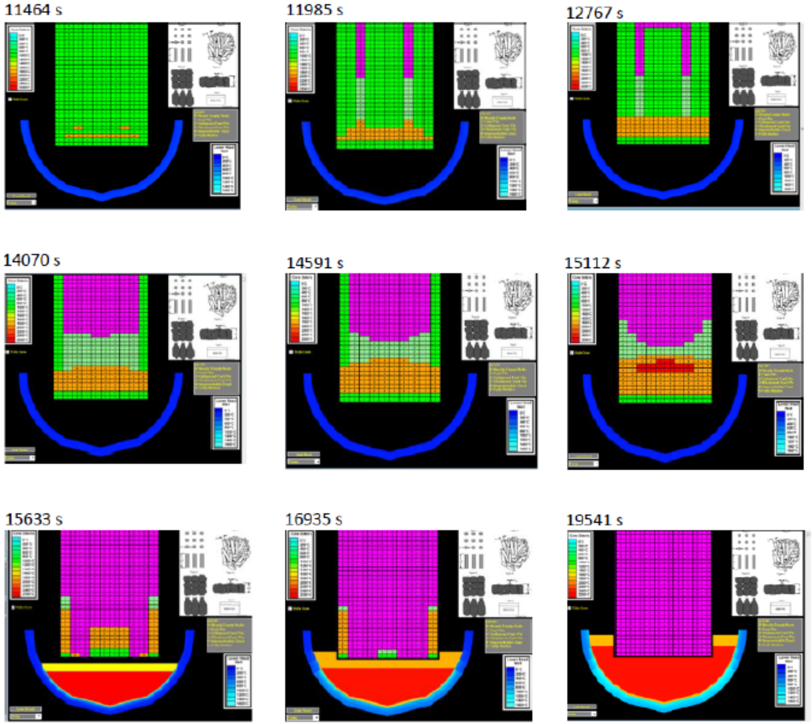


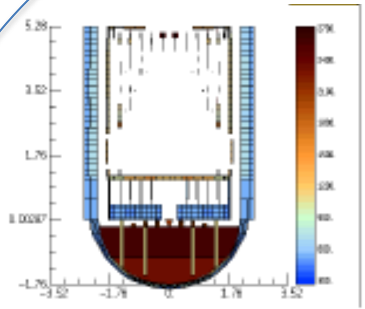
MELCOR Core Degradation Representation

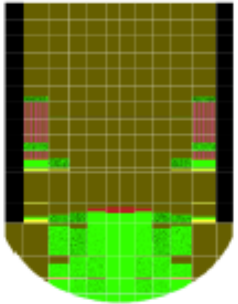


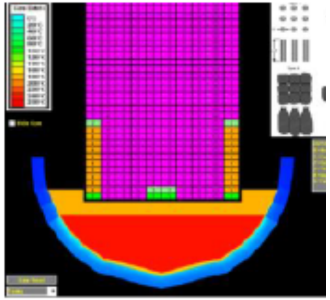
EXAMPLE: Analyses of an Unmitigated Station Blackout Transient in a Generic PWR-900 with ASTEC, MAAP and MELCOR Codes, NUREG/IA-0515

MAAP Core Degradation Representation









SOURCES OF UNCERTAINTY AND METHODOLOGIES

- Providing the result of a best estimate calculation alone may be not sufficient and the evaluation of the uncertainty on the results is required.
- In general the sources of uncertainty can be grouped as:
  - Code uncertainty (e.g. approximations in the conservation equation and in the closure models and correlations)
  - Representation uncertainty (nodalization effect),
  - Scaling issue (codes validated against scaled-down facilities),
  - Plant uncertainty (e.g. initial and boundary conditions),
  - User effect.
- Several methodologies have been developed in the past to perform uncertainty analyses. In particular these uncertainty methodologies can be grouped in:
  - Methods to propagate input uncertainty, divided in
    - Probabilistic (e.g. CSAU, GES. IPSN, etc.)
    - Deterministic methods (e.g. AEAW, EDF-Framatome, etc.);
  - Method to extrapolate output uncertainty (e.g. UMAE).

SEVERE ACCIDENT AND UNCERTAINTY ESTIMATION, CURRENT ACTIVITIES

- Considering:
  - The need to reduce and/or evaluate some uncertainties still present and
  - The reached level of development and maturity of SA codes and their application in the assessment of SAMG,

  the discussion and application of SA progression analyses with uncertainty estimation is currently a key topic in the BEPU framework.

- Considering the key role of SA code for deterministic safety analyses and source term evaluations, several research activities in national and international frameworks are in progress and are planned to reduce and/or estimate the uncertainty in SA phenomena prediction.

- In this framework two main activities are currently in progress:
  - Management and Uncertainties Of Severe Accidents (MUSA) project, founded in HORIZON 2020 EURATOM NFRP-2018 call - Safety assessments to improve accident management strategies for Generation II & III reactors;
  - IAEA CRP on “Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water Cooled Reactors (I31033)

MANAGEMENT AND UNCERTAINTIES OF SEVERE ACCIDENTS (MUSA)

- Management and Uncertainties Of Severe Accidents (MUSA):
  - It was founded in Horizon 2020 EURATOM NFRP-2018 call on "Safety assessments to improve accident management strategies for generation II and III reactors". **MUSA GA 847441;**
  - It aims to establish a harmonised approach for the analysis of uncertainties and sensitivities associated with severe accident (SA) analysis among EU and non-EU entities;
  - It was launched in June 2019;
  - It is coordinated by Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (CIEMAT) in Madrid, Spain
  - It has the NUGENIA label that recognises the excellence of the project proposal (obtained on 7 July 2018)

- MUSA in Numbers
  - 48 months;
  - Budget of € 5,768,452.50;
  - 28 partners (The MUSA project includes partnerships with non-European institutions (Canada, China, Japan, South Korea and USA).
  - 16 countries

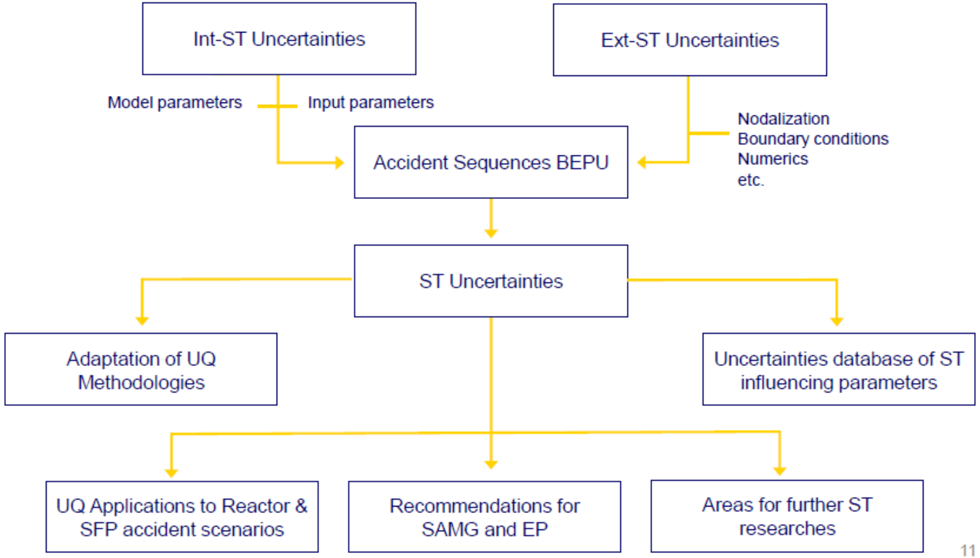
MUSA – OBJECTIVE AND SCOPE

- OBJECTIVES: To assess the capability of SA codes when modelling accident scenarios:
  - Identification and characterization of input & models uncertainties;
  - Assessment of available UASA methodologies;
  - Adaptation of available UASA methodologies;
  - Application to postulated NPP scenarios.

- SCOPE:
  - Gen II, Gen III & Gen III+;
  - Reactor & SFP;
  - Focused on Source Term;
  - SA measures (existing & innovative).

- WEBSITE
  - http://musa-h2020.eu/

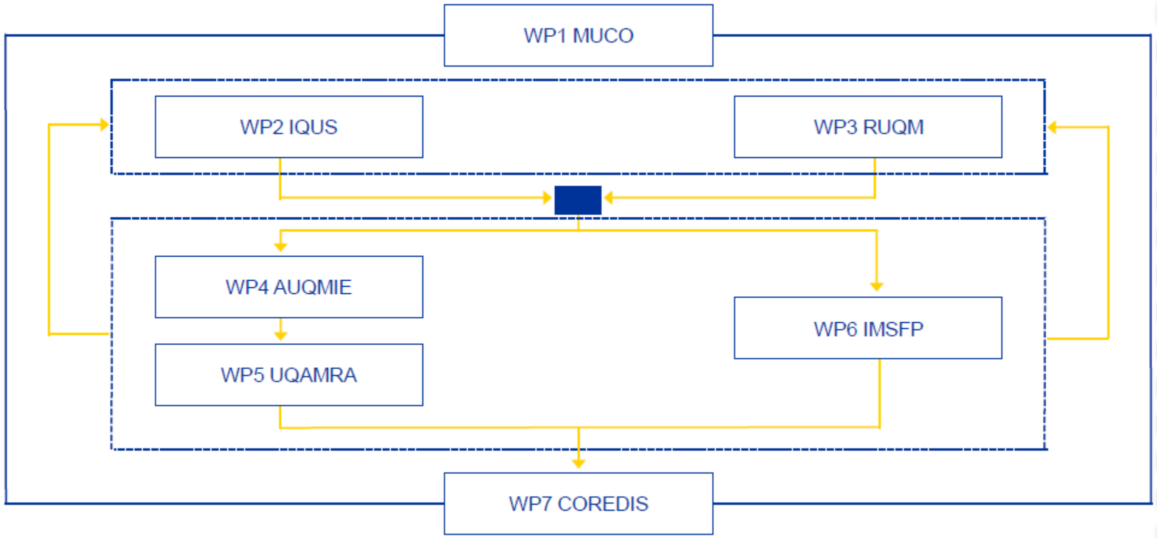
MUSA – APPROACH AND SPECIFIC IMPACT



SPECIFIC IMPACT:

- A systematic assessment of the uncertainty band affecting ST in risk dominant sequences;
- Guidelines to systematic conduct BEPU analysis in the SA domain;
- A database with the characterization (upper and lower bound and pdf) of uncertainties in input deck parameters.
- Insights into key elements affecting SAM implementation (i.e., timing);
- Additional means and actions that might optimize the accident management, both in reactors and SFPs;
- Hands-on training & identification of major challenges.

MUSA – FUNCTIONAL STRUCTURE AND GENERIC EXPECTED OUTCOMES



- **<u>GENERIC EXPECTED OUTCOMES</u>**

  - Close-out open issues in the SA area: uncertainties governing the Source Term (ST) estimates will be identified so that future research can reduce ST predictions uncertainties.
  - Increase safety margins of power plants under operation (support to NPP assessments).
  - Improve emergency response measures and SAM strategies.
  - Enhance nuclear safety while boosting the EU safety requirements' implementation.

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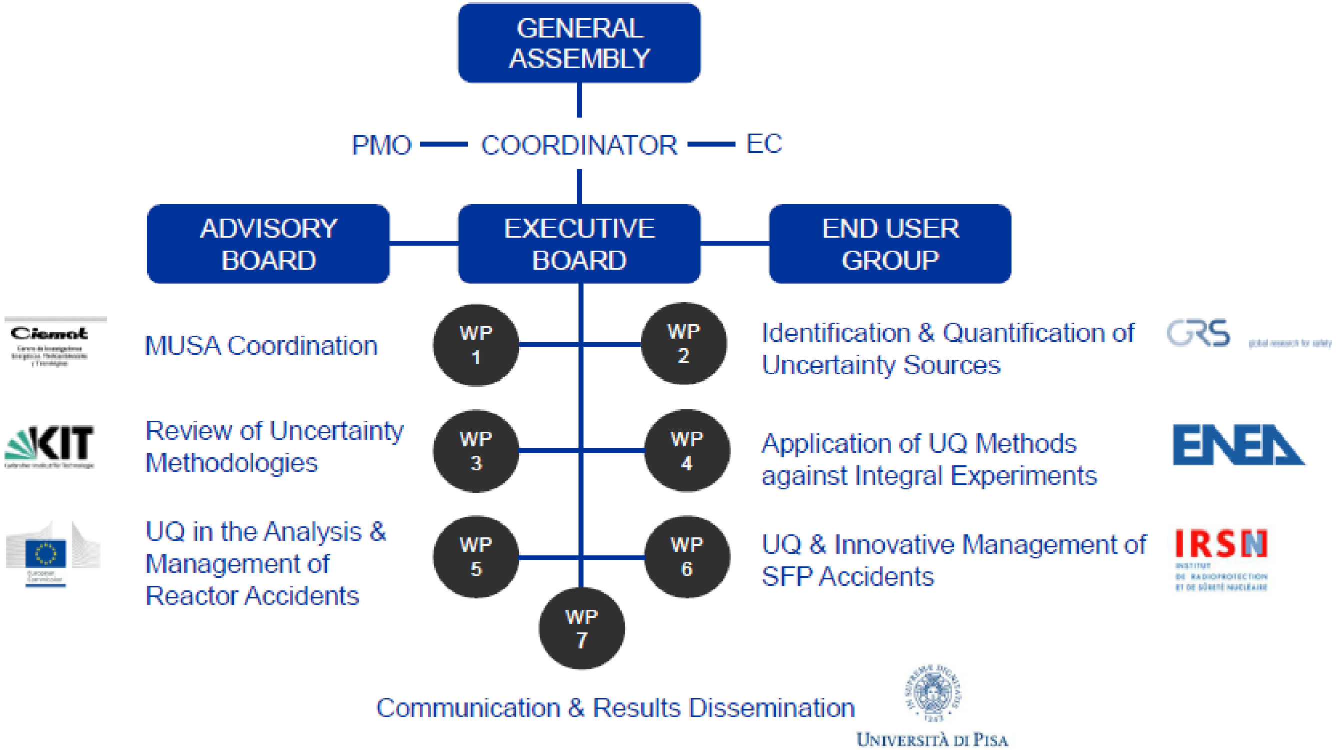
MANAGEMENT AND UNCERTAINTIES OF SEVERE ACCIDENTS (MUSA) – PROJECT GOVERNANCE



MUSA – CONSORTIUM MEMBERS

MUSA – STATUS

- MUSA was approved by EC and launched on 1st June 2019.
- WPs on Identification and quantification of input uncertainties (WP2) and on UQ methodologies (WP3) working at full power.
- Training with UQ methodologies on FPT1 has just started (BE calculation).
- Application packages (WP5 & WP6) will be launched in Sept. 2020.

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

IAEA-CRP: ADVANCING THE STATE-OF-PRACTICE IN UNCERTAINTY AND SENSITIVITY METHODOLOGIES FOR SEVERE ACCIDENT ANALYSIS IN WATER COOLED REACTORS (I31033)

- *CRP INITIATION: 2017 IAEA Technical Meeting :*
  - Common observations from 2017 Technical Meeting led to the CRP rationale
    - Severe accident codes embody complex multi discipline physics spanning wide range of phenomena often outside user range of experience and competency;
    - Code users are often unsure about correctness or accuracy of their plant accident analyses;
    - Code users are often not aware of importance or impact of uncertainty and variability in predicted code results.
  - This CRP is aimed at improving the state of practice in severe accident analyses by examining and characterizing the impact of uncertainty and variability on severe accident analyses

Uncertainty analyses (uncertainty methodology and tools)	STATUS: — Some uncertainty analyses are in progress and relevant examples are available in the public international scientific technical literature; — It should be a common practice in research framework. NEED: — Elaboration of common practices for performing sensitivity and uncertainty analyses; — Automatic coupling between uncertainty tools and codes; — Which is the recent approach that we should follow when we do severe accident analyses (sensitivity and uncertainty); — Express needs for utilities to do **uncertainty analysis** and do it properly and affordably. Recommendations to IAEA: **Plan a next meeting about the use of uncertainty in severe accident analysis — not how to do UA, but how uncertainty is handled in severe accident analysis and severe accident response training** (this could be coupled with other consistent initiative in other framework) **Develop a new CRP and benchmarking of the codes in this area.**

- Detail information about the TM "Technical Meeting on the Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors IAEA Headquarters Vienna, Austria, 9–12 October 2017" can be found in IAEA TECDOC 1872: Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors: https://www-pub.iaea.org/MTCD/Publications/PDF/TE-1872web.pdf

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IAEA-CRP: ADVANCING THE STATE-OF-PRACTICE IN UNCERTAINTY AND SENSITIVITY METHODOLOGIES FOR SEVERE ACCIDENT ANALYSIS IN WATER COOLED REACTORS (I31033)

- **IAEA CRP TITLED:**
  - Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water Cooled Reactors

- **Website:**
  - https://www.iaea.org/newscenter/news/new-crp-advancing-the-state-of-practice-in-uncertainty-and-sensitivity-methodologies-for-severe-accident-analysis-in-water-cooled-reactors-i31033

- **Duration:**
  - 5 years
  - Start date: 7 June 2019
  - Expected end Date 6 June 2024

- **CRP Objectives**
  - Achieve significant improvement in sophistication and quality of severe accident analyses performed by the participants from Member States with well developed knowledge, adequate simulation capabilities (both software and hardware) and long years of relevant practice;
  - Foster national excellence and international cooperation through an exercise to elevate the ability and sophistication of global severe accident code users and participation in benchmark calculations;
  - Share the research results relevant to evaluation of uncertainties in severe accident various codes to contribute to capacity building in developing countries.

IAEA-CRP: ADVANCING THE STATE-OF-PRACTICE IN UNCERTAINTY AND SENSITIVITY METHODOLOGIES FOR SEVERE ACCIDENT ANALYSIS IN WATER COOLED REACTORS (I31033)

- **CRP Outcomes:**
  - Improve capabilities and expertise in Member States to perform state-of-the-art uncertainty and sensitivity analysis with severe accident codes;
  - More defensible application of severe accident codes;
  - Establish best practise for uncertainty and sensitivity analyses in the realm of severe accident analysis;
  - Increase depth and breadth of severe accident analysis and uncertainty/sensitivity analysis using integral severe accident codes by Member States;
  - Elevate ability and sophistication of global severe accident code users with improved characterization of the effect of various sources of uncertainty and variability in the predictive output of relevant codes for advanced WCRs;
  - Foster a common understanding of uncertainty and sensitivity methodologies and tools among Member States.
- **CRP Outputs**
  - IAEA NES on state of practice with lessons learned on best practices in uncertainty and sensitivity methodologies for the severe accidents analyses in WCRs;
  - IAEA TECDOC on uncertainty methods and tools for severe accidents codes with relevant benchmark results
  - Relevant training workshops and courses and supporting lecture materials to be published as the IAEA Training Series Documentations
  - Publications in conference proceedings and peer reviewed journals
  - PhD training programme to strengthen promotion of research on severe accidents simulation and modelling in developing Member States through pair building between agreement holders and contract holders institutes

CRP – IAEA -Participants

- *Argentina : National Atomic Energy Commission (CNEA);*
- *Canada: Canadian Nuclear Laboratory (CNL);*
- *China : Shangai Jiao Tong University (*SJTU)*;*
- *Egypt: Egyptian Nuclear and Radiological Regulatory Authority (ENRRA);*
- *Germany: Karlsuhe Institute of Technology (KIT); Institute for Neutron Physics and Reactor Technology (IKIT- NR);*
- *Ghana: Ghana Atomic Energy Commission (GAEC);*
- *Italy: Italian Agency for New Technology, Energy and Sustainable Economic Development (ENEA);*
- *Lithuania: Lithuanian Energy Institute (LEI);*
- *Malaysia:* Malaysian Nuclear Agency (*MNA- joined after the start of the CRP);*
- *Mexico: National Institute for Nuclear Studies (Instituto Nacional de Investigaciones Nucleares) (*ININ);
- *Pakistan: Pakistan Atomic Energy Commision (PAEC);*
- *Republic of Korea: Korea Atomic Energy Research Institute (KAERI),*
- *Romania: Politechnica University of Bucharest;*
- *Russian Federation: Nuclear Safety Institute of the Russian Academy of Science (IBRAE), OKB "Gidropress", NRC "Kurchatov Institute";*
- *Spain:* Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (*CIEMAT),* Energy Software SLP (*ENSO) (joined after the start of the CRP);*
- *Ukraine: Scientific and technical Center of SE NNEGC Energoatom;*
- *United Arab Emirates: U Sharjah (join after the start of the CRP);*
- *USA: Sandia National Laboratory (SNL) ,* Innovative Systems Software (ISS, *joined after the start of the CRP).*

CRP – IAEA TASKS

- TASK 1: Quench 6 test application uncertainty exercise – LEAD-KIT
- TASK 2: Plant application uncertainty- LEAD-ENEA
  - BWR Mark I – LEAD SNL
  - BWR Mark II – LEAD ININ
  - PWR – LEAD KAERI
  - VVER – LEAD IBRAE
  - SMR – LEAD CNEA
  - CANDU –LEAD CNL

- TASK 3: Develop reports on uncertainty and sensitivity in severe accident analysis
  - TASK 3.1: development of IAEA-TECDOC:
  - In this document will be described all CRP Member uncertainty applications and the lesson learned;
  - TASK 3.2: development IAEA- Nuclear Energy Series document:

    In this document will be reported the critical analyses of the CRP Member uncertainty applications and the best practices.

- TASK 4: Development of PhD training programme:

  It will be developed and implemented for students from countries embarking on nuclear power, in which PhD candidates will be supported by organizations in experienced Member States.

CRP – STATUS

- Along the first year of activity:
  - Each plant application participant has defined the framework of the analyses, its main targets, the figure of merit and performed the reference scenario calculations.
  - Each participant has investigated the uncertainty methodology together with a first draft list of input uncertainty parameters to be studied in the uncertainty exercise that will be developed along the second year of activity.
  - In relation to the Quench exercise the main facility information and experimental data has been distributed together with the input uncertainty parameters and figure of merit of the analyses; each participant performed the reference calculation.

CONCLUSIONS

- Considering the complexity and mutual different interacting and interrelated phenomena/processes along a severe accident transient progression, a key role is played by the state-of-the-art severe accident integral codes.
- The analyses of the current state-of-the-art shows that in the analyses of SA
  - There is a need to reduce some uncertainties still present and
  - A consequent investigation of phenomena/processes, to date not investigated in detail in geometric prototypical experimental facility with prototypical material, should be addressed.
- Considering the key role of SA code for deterministic safety analyses and source term evaluations, several research activities in national and international frameworks are in progress and are planned to reduce and/or estimate the uncertainty in SA phenomena prediction.
- In this framework two main activities are currently in progress:
  - Management and Uncertainties Of Severe Accidents (MUSA) project, founded in HORIZON 2020 EURATOM NFRP-2018 call;
  - IAEA CRP on “Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water Cooled Reactors (I31033)
- These activities are complementary and will produce a important feedback on the international technical nuclear community.

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