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

MAIN R&D NEEDS FOR LW-SMR

MDEP Brainstorming Workshop on Light-Water Small, Medium, and Modular Reactors (LW-SMMR), June 10-11, 2024, in Ankara, Türkiye

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INTRODUCTION

- Passive systems are adopted in NPP since the beginning (e.g. accumulator);
- The Chernobyl and the Fukushima Daiichi events determined an increase of interest in accident mitigation strategy based on the use of passive systems;
- In the last decades, the international technical community, taking into account the operational experience of the current nuclear reactors, started the development of new advanced reactor designs to satisfy:
  - demands of the people to improve the inherent safety of nuclear power plants;
  - demands of the utilities to improve the economic efficiency and reduce the capital costs.
- New passive system concepts have been designed and nowadays are more and more considered among the features desired in future advanced plants in order to increase the inherent safety of the plants.
- Passive safety systems are currently considered in large scale Generation III+ reactors and in advanced Small Modular Reactor (SMR);
- SMR specific features, strengthen the suitability of passive safety systems to reinforce the first three DiD levels: e.g. Lower core power; Integral design of the primary system; Large core surface-to-volume and coolant inventory-to-power ratios; Fuel design.

PASSIVE SYSTEMS NEEDS

- Two interrelated needs on passive systems in general and for SMRs specifically:
  - Safety assessment:
    - Reliability of passive systems;
    - Deterministic safety analyses.
  - Qualification of computational tools including metamodels.

- Deterministic analysis codes: key elements used to develop safety analyses:
  - Results have to be properly qualified;
  - Uncertainty of the results should be quantified;
  - Qualification highly relies on experimental support within the range of application.

- Four main specific subtopics/needs have been identified considering the current State-of-Art:
  - Experimental assessment database;
  - Code modeling;
  - System reliability;
  - System designs and engineering process.

INTRODUCTION

- Passive and, in general, other advanced reactor designs as SMR are characterized by:
  - Some common features with the current generation of reactors (e.g. large scale PWR and BWR) :
    - advanced designs may be characterized by a different ranking of some phenomena.
  - Other features typical of their design, that can be grouped as
    - a) Containment process and interactions with the RCS;
    - b) Low pressure phenomena;
    - c) Phenomena related specifically to new system components or reactor configurations.

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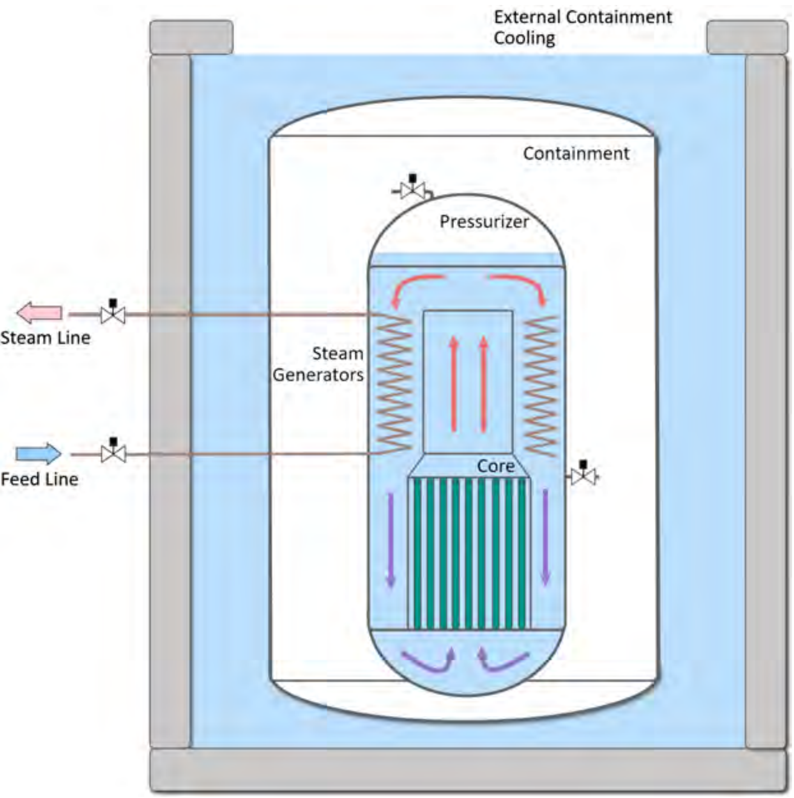
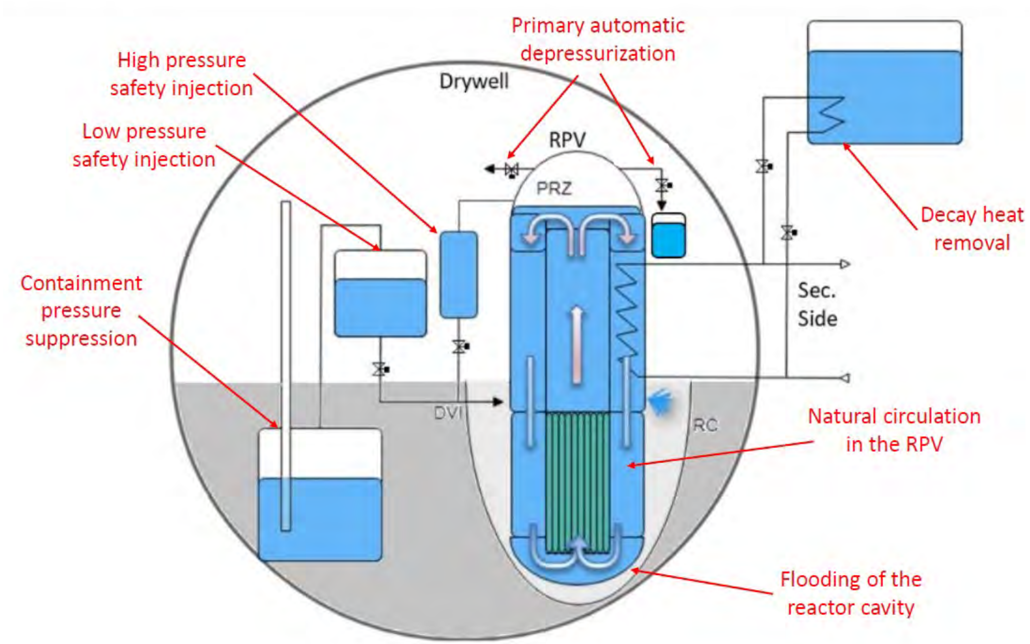
INTRODUCTION

Example of iPWR design with submerged metal containment



Example of iPWR design using several passive systems with a large dry containment



EXPERIMENTAL CAMPAIGN FOR ADAVANCED DESIGN

- Considering that advanced designs are characterized by new kinds of phenomena and accident scenarios, and by a possible different ranking in the phenomena with respect to current reactor designs,
  - further experimental investigations, in SETF and ITF are necessary to extend the "assessment database".

- Considering the current deployment of large-scale advanced reactors using passive systems, such as AP1000, and the possible deployment of advanced Small Modular Reactor (SMR) designs, in the last decades
  - several experimental activities have been developed in several countries for the TH characterization of advanced designs and for the development of experimental data useful for code assessment.

EXPERIMENTAL CAMPAIGN FOR ADVANCED DESIGN SOME EXAMPLES OF IETF

- Several IETFs have been used to investigate natural-circulation phenomena and the thermal-hydraulic response of passive safety systems in advanced reactor designs

Facility	Reference Reactor	Scaling Approach	Scaling Method	Counterpart /similar Test	ICSP-ISP
PWR PACTEL	EPR Like-4L	RPw, RPr, NNC, RNL, S-W			
ATLAS	APR1400 - 2L (1HL-2CL)	TNP, RH, RPw, FPr, NNC, ELN, ECLN, S-W	Three-level scaling	YES	
SNUF	APR1400 - 2L (1HL-2CL)	RH, RPw, RPr, NNC, ELN, ECLN, S-W	Three-level scaling		
APEX	AP600-2L (1HL-2CL)	TNP, RH, RPw, RPr, NNC, ELN, ECLN, S-W	H2TS	YES	
SPES-2	AP600-2L (1HL-2CL)	TP, FH, FPw, FPr, NNC, ELN, ECLN, S-W	P-to-V (Kv) Modification of SPES	CT, ST	
ROSA-AP600	AP600-2L (1HL-2CL)	TP, FH, RPw, FPr, NNC, ELN, NCLN, S-W	P-to-V (Kv) Mmodification of LSTF	YES	
OSU-MASLWR	MASLWR -IWCR	TP, RH, FPw, FPr, NNC, I, S-W-A	H2TS		IAEA ICSP
VISTA-ITL	SMART-IWCR	TNP, RH, FPw, FPr, NNC, S-W	Three-level scaling		
FESTA	SMART-IWCR	TP, RPw, FPr, NNC, S-W	Three level scaling		
IST	mPOWER IWCR	TP, FH, FPr, NNC, S-W			

OSU MASLWR: Upper region of the hot leg riser has an OD equal to 114.3 mm;
NIST: A modification of the OSU MASLWR facility and used for the simulation of NUSCALE IWCR and TP, RH, FPw, FPr, NNC, I, S W A are the main characteristics

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EXPERIMENTAL CAMPAIGN FOR ADAVANCED DESIGN SOME EXAMPLES OF SETF

- In relation to SETF, several facilities have been constructed to test various systems and components of current and advanced reactors.

  - Considering NuScale design, for example, two experimental activities have been conducted at SIET (Italy) to characterize the TH behavior and structural dynamic conditions of the helical coil SG.
    - The two experimental campaigns use full length SG helical tubes with high pressure and temperature.

  - Considering SMART design some SETF have been constructed to test different phenomena.
    - SCOP facility (SMART Core flow distribution and Pressure drop test facility) was adopted to verify the core inlet flow rate and pressure distributions;
    - SWAT facility (SMART ECC Water Asymmetric Two-phase Choking Test Facility) was adopted to test ECC bypass and
    - FTHEL facility (Freon Thermal Hydraulic Experimental Loop) was used to derive a database for CHF and to verify the DNBR model in the safety analyses and the core-design codes adopting a 5x5 rod bundle.

- In relation to passive systems, PERSEO is a full-scale SETF-component test aimed at studying a new passive decay heat removal system operating in natural circulation. It was built at SIET (Italy) modifying the existing PANTHERS IC-PCC facility.

EXPERIMENTAL DATABASE GAPS AND NEEDS

GAPS:

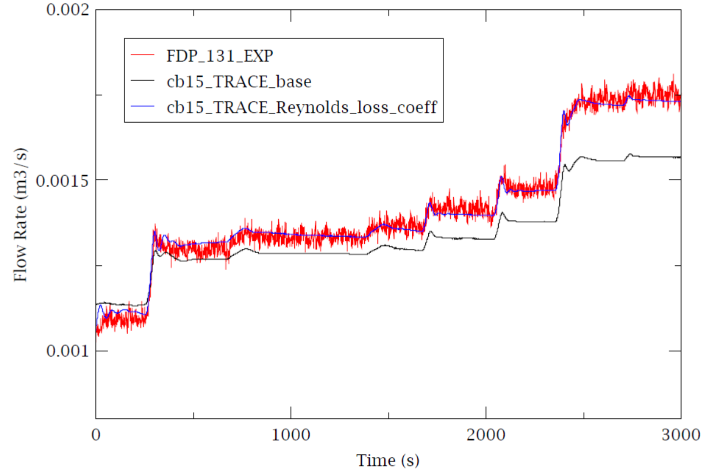
- Even if some large experiments in the world have been done for characterizing the thermal-hydraulic phenomena of passive system, only few experimental data have been currently used by the international community.
- There is a lack of available experimental test campaigns on which we can rely for an exhaustive evaluation of the State-Of-Art thermal-hydraulic codes (some nuclear actors have built their own test facilities, but the results are not public).

NEEDS:

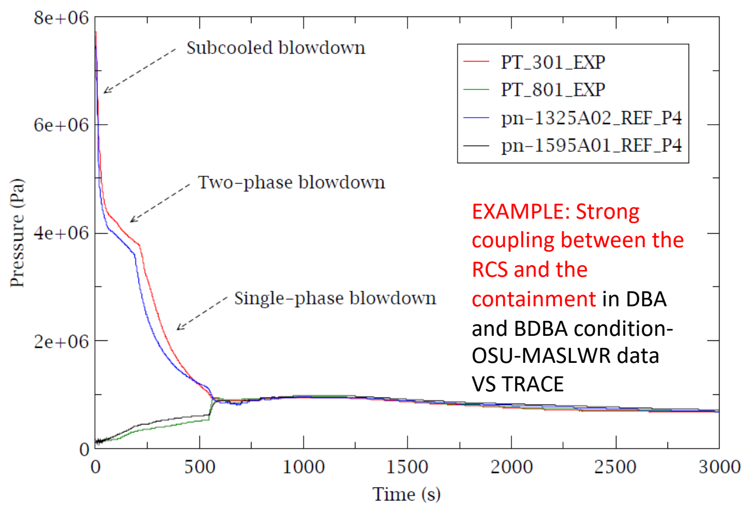
- Large scale facilities (ITF and SETF) characterized by low uncertainty measurement at low flow regime;
- Scaling issue should be addressed;
- Experiments characterized by well-instrumented tests for validating CFD in relation to 3D phenomena (e.g. mixing with buoyancy effects);
- Produce high-resolution data still needed to advance the fundamental understanding of phenomena (e.g. flow boiling and two-phase flow, conjugate heat transfer, …..) typically relevant for passive systems in advanced reactor concepts;
- Dedicated large scale facilities will be needed to evaluate the capability of the codes to accurately reproduce:
  - Integral configurations and passive system loops (e.g. pressure drop at different mass flow rate, etc);
  - Strong coupling between the RCS and the containment;
  - 3D phenomena.

NEEDS FOR SMR MODELING

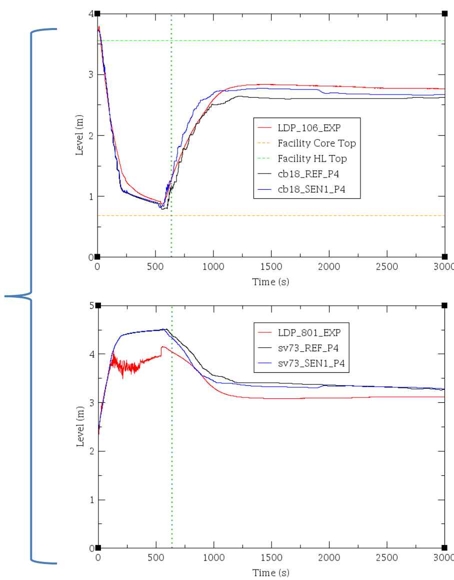
- **Identify current modeling limitations:**
  - Codes need to be proven able to accurately predict the T-H phenomena typical of advanced designs as:
    - Integral configuration;
    - Passive mitigation strategy.
  - Review of the model/constitutive equations implemented in the codes (or passive system models already developed but still not implemented in the codes) and their representativeness.



EXAMPLE: Integral configurations and passive system loops (e.g. pressure drops at different mass flow rate, etc)- OSU-MASLWR data VS TRACE



EXAMPLE: Strong coupling between the RCS and the containment in DBA and BDBA condition- OSU-MASLWR data VS TRACE



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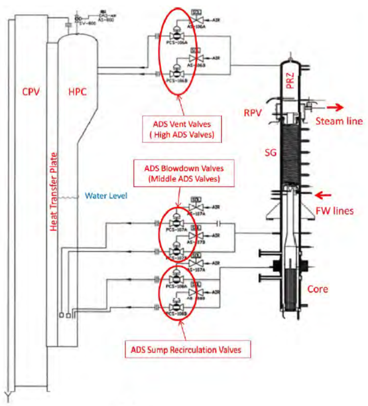
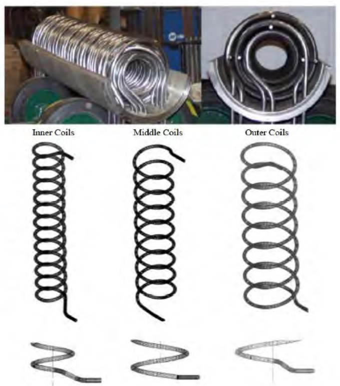
NEEDS FOR SMR MODELING

OSU-MASLWR and Helical Coil Compact SG



a)



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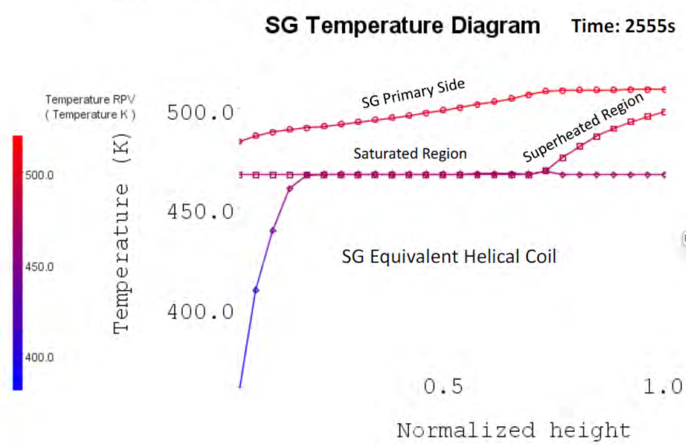
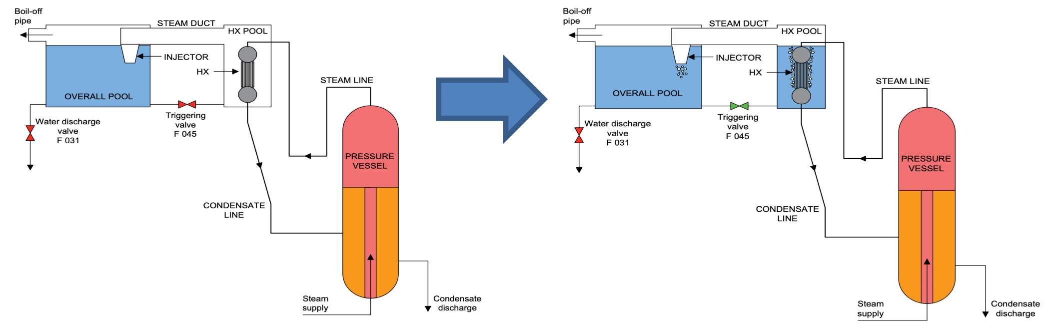
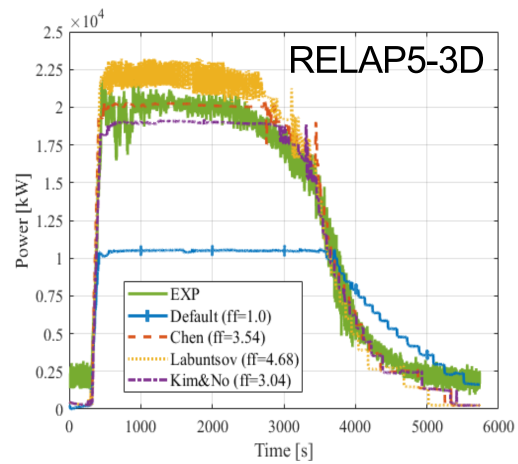


PERSEO and full-scale heat transfer phenomena of in-pool heat exchangers (tube-side and pool-side)





- In PhW2 of Test 7 Part 2, the computed exchanged power is nearly half of the experimental one (around 10 MW instead of 20 MW).
- In order to correctly predict the heat transfer, and the transient progression, a correction factor for the condensation heat transfer coefficient is needed.
- After the application of the correction factor ("fouling factor") the code is able to qualitatively and quantitatively predict the heat transfer in the heat exchanger.

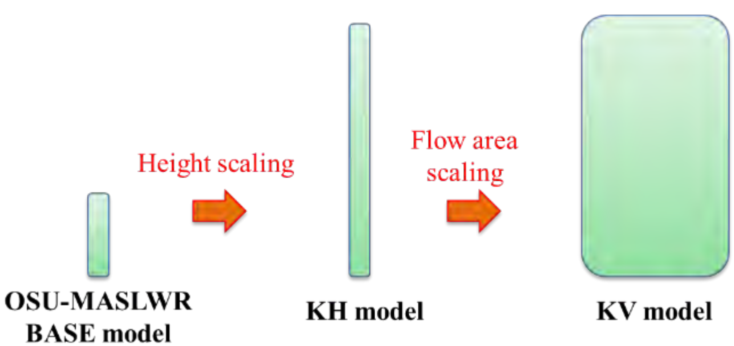
NEEDS FOR SMR MODELING

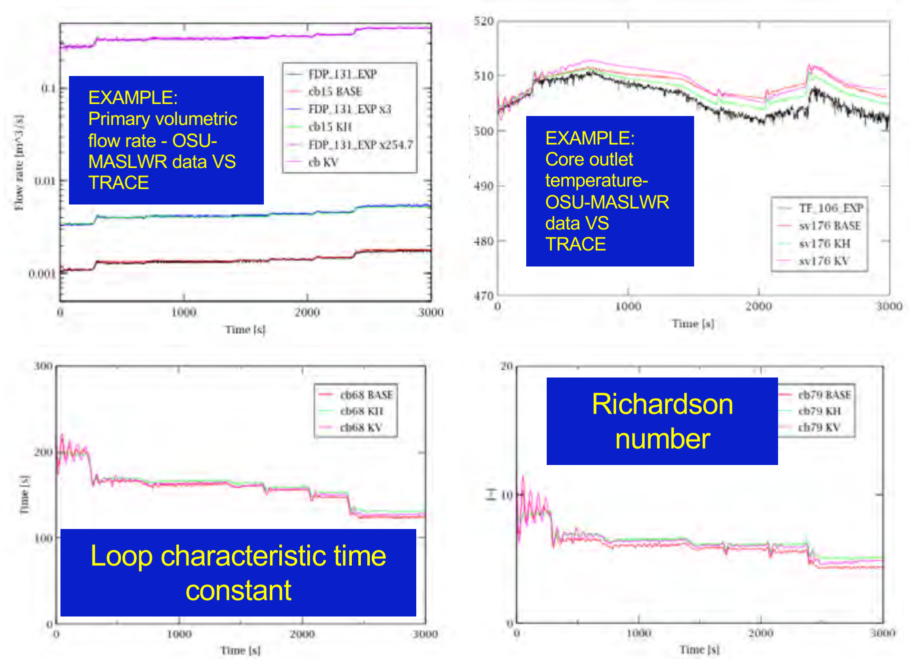
- Major sources of uncertainty in code modelling need to be identified and characterized.

- Scaling issue should be addressed (e.g., validation of the codes with experimental data at different scales).

Example:

assess the scaling-up capability of the OSU-MASLWR nodalization against natural circulation phenomenology ➔ numerical scaling if no counter part test are available





GAPS AND NEEDS FOR SMR MODELING

GAPS:

- State-of-art tools have been evaluated against current operating reactor phenomena
- Validation of the DSA codes for all passive system operation mode will be necessary and relevant for several challenges.
- Some activities have been done or are currently in progress in domestic and international collaborative framework to assess the capability of code for specific passive system phenomena.
- Code modelling uncertainty can affect the predicted reliability

NEEDS:

- Efforts should be still made to exhaustive validate state-of art codes with the specificities of
  - Passive systems (e.g. low pressure, natural circulation, condensation heat exchange) and
  - SMR (e.g. integral configurations, compact steam generator, etc)
- Develop specific models for new reactor configurations and components;
- Scaling issue should be addressed;
- Major sources of uncertainty in code modelling need to be identified and characterized.

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RELIABILITY EVALUATION OF PASSIVE SYSTEM

- The thermal-hydraulic behaviour of passive safety system is determined by the natural circulation characteristics in both nominal operation and transient conditions.
- Possible failures, or deviations from the working conditions during transient and design specifications, may occur;
- Analysis of the thermal-hydraulic phenomena that may occur in the passive safety system by using best estimate thermal hydraulic system codes is necessary to assess the performance;
- Reliability of passive safety system is linked with the performance of the system and its evaluation is strongly connected with the code uncertainties:
  - large uncertainty bands, i.e., due to modelling limitations in the state-of-art T/H system codes➔ analysis results may include large un-reliability values that may not represent true characteristics of the passive safety system.
- Concept and method of uncertainty evaluation, and the qualification and usefulness of the method, shall be positively demonstrated.
- Same passive safety system can be involved in the different levels of DiD and, as a consequence, the same phenomenology can be the basis of systems acting into two different DiD levels.
  - functional failure could be a potential common cause failure.

CURRENT STATUS

- Passive systems are typically found satisfactorily to accomplish the target mission when one or few operational scenarios are investigated;
- When a large spectrum of scenarios, in the order of thousands, is analyzed, failures are found in relation to target mission (e.g. REPAS and RMPS application at the beginning of 2000):
    - Challenging➔ determining the probability for the occurrence of each failed-scenario.
- It is not practically possible to perform in scaled-down experimental test facilities the needed number of experiments (order of thousands);
    - It is possible to develop an equivalent number of numerical simulations:
        - Challenging:
            - Extremely expensive
            - Requires a full qualification of the simulation tools.
- Key role that TH system codes have in evaluating the reliability of passive safety system:
    - Challenging:
        - Calculated results could be questionable also for considering the structure and the capabilities of those codes (e.g. user effect and selection of pressure drop coefficients)

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RELIABILITY ANALYSES GAPS AND NEEDS

GAPS

- Functional failures are addressed by advanced reactors designers but must be considered also in an independent safety review process
- In relation with deterministic safety demonstration:
  - Guidance on requirements specific to passive systems and their features (activation, no external power, …);
  - Guidance on the methodologies appropriate to model the system failure modes.

NEEDS

- Identify and review the methodologies currently used for reliability evaluation (e.g., REPAS, RMPS, APSRA);
- Assess functional failure related to the T-H phenomena driving the operation of the systems and assess the related uncertainties;
- Reliability region of passive systems needs to be investigated and uncertainties should be considered;
- Define requirements and appropriate methodologies to model the system behaviour and its dependencies to the variation of accident conditions, without over-conservatism (including aggravating events);
- Reduce the width of uncertainty bands on the key parameters driving the physics of passive systems in transient conditions;
- Characterize the entire spectrum of T-H conditions, that can take place along a transient, and that can affect the passive system target safety function fulfillment.

CONCLUSIONS

- ❑ Passive systems are being adopted in advanced reactor designs.
- ❑ Passive safety system are characterized by Opportunity and Challenges;
- ❑ Currently reactor designers have been successful in addressing the challenges, and several Countries support the use of passive safety systems;
- ❑ Evaluation of passive safety system is recognized as complex task:
  - o Performance of passive system is often based on small scaled-down experimental test facilities leading to uncertainty in their performance.
  - o Uncertainties must be considered and accounted for, and methods are available to do it.
- ❑ Some Countries have independently verified the passive safety systems features in the advanced designs presented;
- ❑ A comprehensive evaluation and review of the design, supported by sufficient experimental data, is necessary for the determination of safety adequacy of passive safety systems.

CONCLUSIONS

- Some methodologies have been developed specifically to evaluate the reliability of passive systems (e.g. REPAS, RMPS).

- Some challenges are present in the reliability evaluation of passive systems, for example:
  - It is not practically possible to perform in scaled-down experimental test facilities the needed number of experiments (order of thousands).
  - Validation of thermal-hydraulic system codes against the phenomena typical of passive systems.
  - Large number or code runs usually needed to obtain the convergence of the value of the figures of merit selected.
  - Quantification of code uncertainties within the evaluation of the reliability region.

CONCLUSIONS

- ❑ Define the needed experiments and enlarge the database for code validation.
- ❑ Scaling issue should be addressed and major sources of uncertainty in code modelling need to be identified and characterized.
- ❑ Apply the available methodologies for passive system reliability evaluation to compare different systems with the same target mission;
- ❑ Reliability region of passive systems needs to be investigated and uncertainties should be considered;
- ❑ Define requirements and appropriate methodologies to model the system behaviour and its dependencies to the variation of accident conditions, without over-conservatism (including aggravating events);
- ❑ Evaluate the adoption of surrogate modeling to reduce the computational cost of the large number of simulations needed for reliability evaluation (this adds the challenge of surrogate modeling validation and related uncertainty quantification).

SPECIFIC CONCLUSIONS

- Specific models for the new components available in a SMR should be implemented in the codes (e.g. modeling for compact SG, specific passive system modeling, etc.);

- The models implemented in the code should be validated against full scale SEF data;
  - Envelopment of full scale SETF that minimizes the distortion in BIC;

- The current capabilities of computer codes should be benchmarked against the available assessment database regarding SMR and passive system ITF, and SETF.
  - Validation against counterpart/similar test is also suggested;
  - New international benchmark activity should be launched.

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SPECIFIC CONCLUSIONS

- In relation to the Integral RPV configuration, the following points should be considered:
  - Throughout the nodalization development, accurate characterization of the form losses along the integral RPV (geometric discontinuities, RPV internal structures, etc.) shall constitute a key modelling aspect;
  - Accurate characterization of the form losses is true also for the modelling of the passive system circuit (e.g. EHRS circuit);

- The validation of computational tools in order to characterize the coupling between the RCS and the containment.

- In order to more accurately predict some phenomena in the containment and passive system, 3D model available in thermal hydraulic system codes (e.g. to simulate mixing phenomena in containment and pool, to capture some 3D thermal–hydraulic detail in the RPV, etc.) should be explored and validated against experimental data.

CONCLUSIONS

- 3D CFD tools become valuable when multi-dimensional effects play an important role, e.g. in single-phase turbulent mixing problems, including temperature mixing, mixing of chemical components in a multi-component mixture (boron in water, hydrogen in gas) and temperature (density) stratification.

  - Two-phase CFD analysis is much less mature than single-phase CFD, but significant progress has been made in the past decade;
  - Multiscale analysis using various numerical tools at different scales will help in the future to provide more accurate and reliable solutions to reactor issues;
  - The simulation capability of details of local phenomena aiming for a replica of the phenomena must be improved;
  - Up-scaling modeling methods should be developed to use small-scale simulations for improving the closure laws used in SYS TH codes;
  - The CFD tools also should follow an appropriate process of code validation to prove their capability for extrapolation to the NPP- prototype phenomena.

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CONCLUSIONS

- Severe accident codes: Need a sufficient accurate modeling of thermal–hydraulic phenomena because the evolution of a SA sequence, involving several different phenomena (e.g. core degradation; fission products chemistry and transport; containment behaviour; etc.), is strongly affected by the thermal-hydraulics evolution of the reactor.

- It is necessary to analyze the capability of the SA codes to predict the main phenomena involved in the passive mitigation strategy of a generic SMR in order to apply a severe accident code for the design and assessment of accident management's strategy in the view of severe accident and emergency plan zone SMR licensing needs:
  - Currently it is ongoing an Horizon Euratom project SASPAM-SA (Safety Analysis of SMR with Passive Mitigation strategies - Severe Accident), funded in HORIZON-EURATOM-2021-NRT-01-01, "Safety of operating nuclear power plants and research reactors", having as:
    - Key Objective:
      - ✓ to investigate the applicability and transfer of the operating large-LWR reactor knowledge and know-how to the near-term deployment of integral PWR (iPWR), in the view of Severe Accident (SA) and Emergency Planning Zone (EPZ) European licensing analyses needs.
    - Key Outcomes:
      - ✓ To be supportive for the iPWR licensing process by bringing up key elements of the safety demonstration needed;
      - ✓ To speed up the licensing and siting process of iPWRs in Europe.

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PHENOMENA AND TH ASPECTS IDENTIFIED FOR CURRENT GENERATION REACTOR

Basic Phenomena	o Evaporation due to depressurization o Evaporation due to heat input o Condensation due to pressurization o Condensation due to heat removal o Interfacial friction in vertical flow o Interfacial friction in horizontal flow o Wall to fluid friction o Pressure drops at geometric discontinuities o Pressure wave propagation
Critical Flow	o Break o Valves o Pipes
Phase Separation/Vertical Flow with and Without 1 Mixture Level	o Pipes/Plena o Core o Downcomer
Stratification in Horizontal Flow	o Pipes
Phase Separation At Branches	o Branches
Entrainment/Deentrainment	o Core o Upper Plenum o Downcomer o Steam Generator Tube o Steam generator mixing chamber (PWR); o Hot leg with ECCI (PWR)
Liquid-Vapour Mixing With 1 Condensation	o Core o Upper Plenum o Downcomer o Steam Generator Tube o Steam Generator Mixing Chamber (PWR) o Hot Leg with ECCI (PWR)
Condensation in Stratified Conditions	o Pressurizer (PWR) o Steam generator primary side (PWR) o Steam generator secondary side (PWR) o Horizontal pipes

Spray Effects		o Core (BWR) o Pressurizer (PWR) o Once-Through Steam Generator Secondary Side (PWR)
Countercurrent Flow / 1 Countercurrent Flow Limitation		o Upper Tie Plate o Channel Inlet Orifices (BWR) o Hot and Cold Leg o Steam Generator Tube (PWR) o Downcomer o Surgeline (PWR)
Global Multidimensional 1 Fluid Temperature, Void 2 And Flow Distribution		o Upper plenum o Core o Downcomer o Steam generator secondary side
Heat Transfer	Natural or Forced Convection Subcooled/Nucleate Boiling DNB/DryoutPost Critical Heat Flux Radiation Condensation	o Core, steam generator, structures o Core, steam generator, structures o Core, steam generator, structures o Core, steam generator, structures o Core o Steam generator structure
Quench Front Propagation/Rewet		o Fuel rods o Channel walls and water rods (BWR)
Lower Plenum Flashing Guide Tube Flashing (BWR) One And Two Phase Impeller-Pump Behaviour One And Two Phase Jet-Pump Behaviour (BWR) Separator Behaviour Steam Dryer Behaviour Accumulator Behaviour Loop Seal Filling And Clearance (PWR) Ecc Bypass/Downcomer Penetration Parallel Channel Instabilities (BWR) Boron Mixing And Transport Non-Condensable Gas Effect (PWR) Lower Plenum Entrainment		

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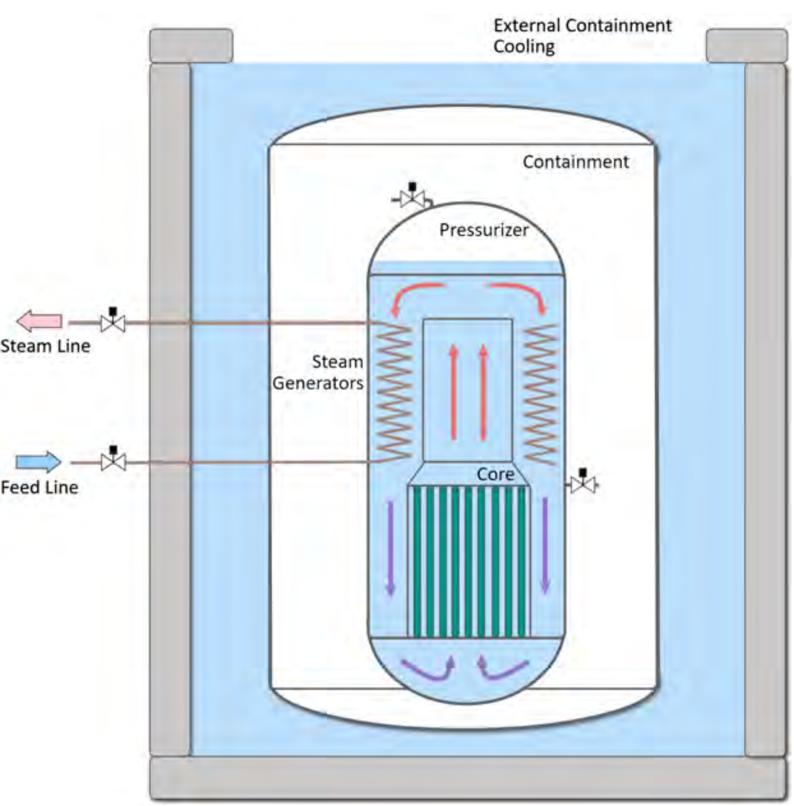
PHENOMENA AND TH ASPECTS FOR REACTOR USING PASSIVE SYSTEMS

PHENOMENA	CHARACTERIZING THERMAL-HYDRAULIC ASPECT
Behaviour in large pools of liquid	o Thermal stratification o Natural/forced convection and circulation o Steam condensation (e.g. chugging, etc.) o Heat and mass transfer at the upper interface (e.g. vaporization) o Liquid draining from small openings (steam and gas transport)
Effects of non-condensable gases on condensation heat transfer	o Effect on mixture to wall heat transfer coefficient o Mixing with liquid phase o Mixing with steam phase o Stratification in large volumes at very low velocities
Condensation on containment structures	o Coupling with conduction in larger structures
Behaviour of containment emergency systems (PCCS, ectrenal Vooling, etc)	o Interaction with primary cooling loops
Thermo-fluid dynamics and pressure drops in various geometrical configurations	o 3-D large flow paths e.g. around open doors and stair wells o connection of big pipes with pools, etc. o Gas liquid phase separation at low Re and in laminar flow o Local pressure drops
Natural circulation	o Interaction among parallel circulation loops inside and outside the vessel o Influence of non-condensable gases o Stability o Reflux condensation
Steam liquid interaction	o Direct condensation o Pressure waves due to condensation
Gravity driven cooling and accumulator behaviour	o Core cooling and core flooding
Liquid temperature stratification	o Lower plenum of vessel o Down-comer of vessel o Horizontal/vertical piping
Behaviour of emergency heat exchangers and isolation condensers	o Low pressure phenomena
Stratification and mixing of boron	o Interaction between chemical and thermo-hydraulic problems o Time delay for the boron to become effective in the core
Core make-up tank behaviour	o Thermal stratification; Natural Circulation

INTRODUCTION

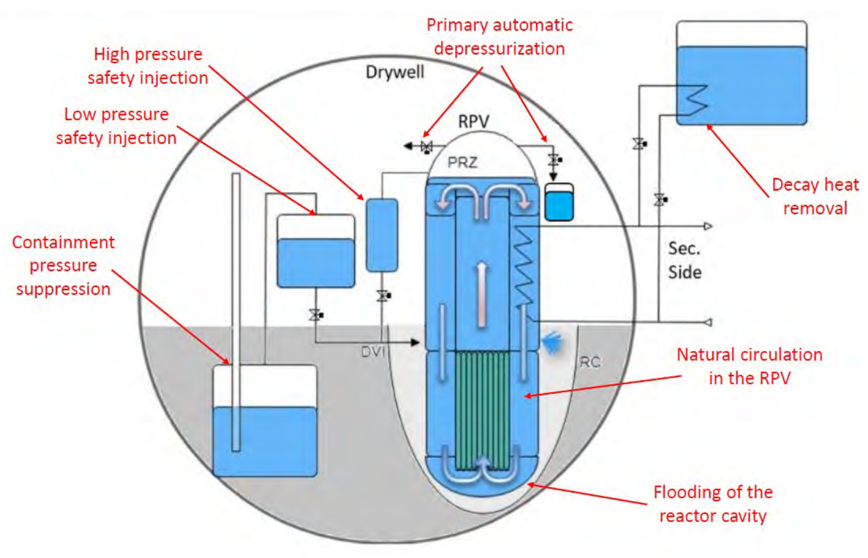
Example of iPWR design with submerged metal containment



- In relation to the containment process and interactions with the RCS (a):
  - It is not possible to consider the RCS as a boundary condition for the containment.
  - it is necessary to consider the behavior of the containment TH coupled with the behavior of the RCS and to characterize the RCS/containment coupled behavior during the transient evolution.
  - Passive mitigation strategies could be based on a natural circulation loop covering both systems to remove the decay heat.

INTRODUCTION

Example of iPWR design using several passive systems with a large dry containment



- In relation to phenomena taking place at low pressure (b), as the atmospheric pressure it can be mentioned:
  - Natural circulation phenomena (e.g. interaction among parallel circulation loops inside and outside the vessel and influence of non-condensable gases),
  - Steam liquid interaction phenomena (e.g. direct condensation),
  - Gravity driven reflood phenomena (e.g. heat transfer coefficient) and
  - Liquid temperature stratification phenomena (e.g. vessel lower plenum and downcomer).

- In relation to the phenomena related specifically to new system components or reactor configurations (c), as example it can be mentioned
  - Natural circulation in integral type configuration (in transient and steady operation),
  - Behavior of compact SG, as helical coiled ones, and
  - Passive residual heat removal systems.

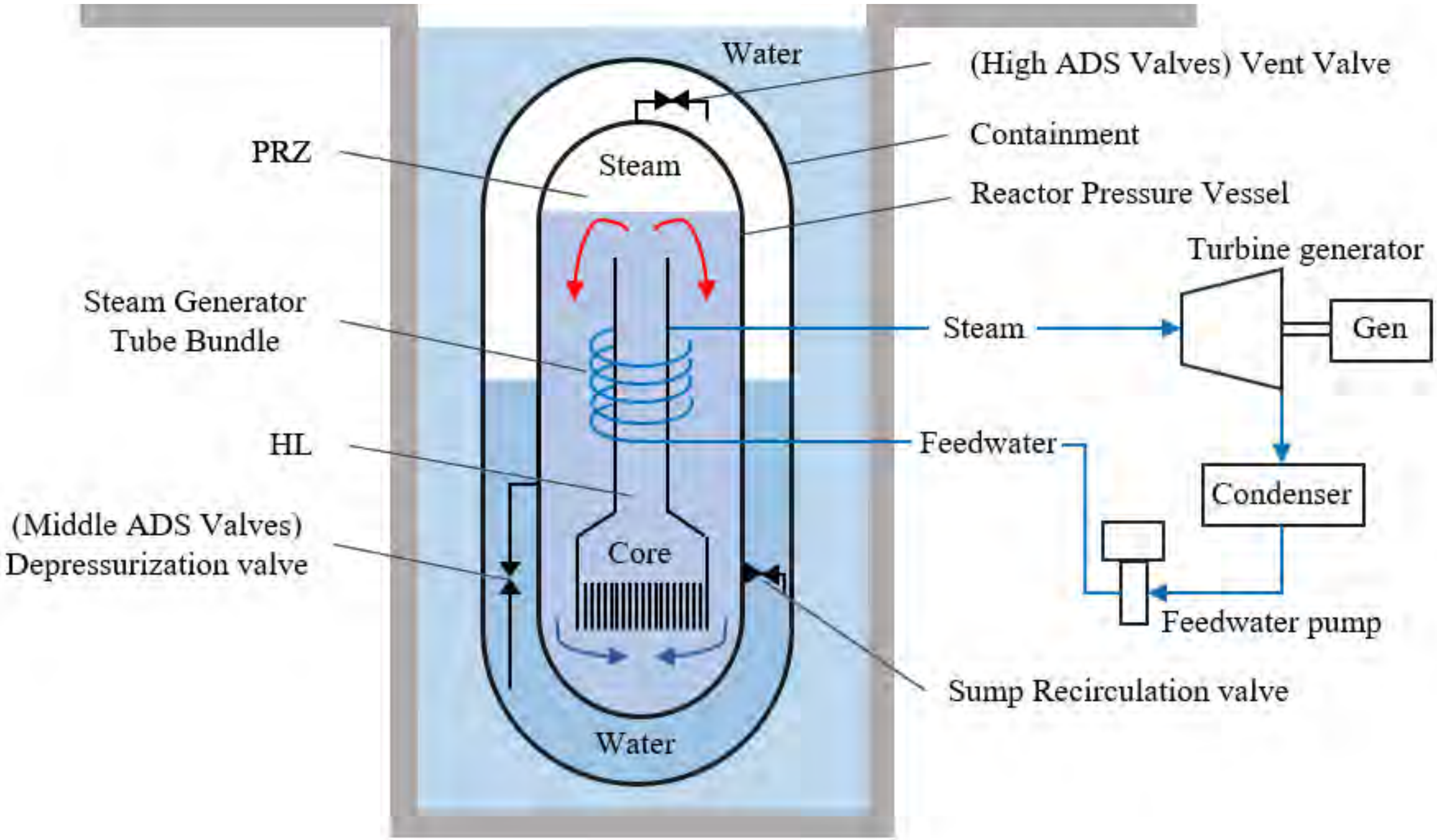
MULTI-APPLICATION SMALL LIGHT WATER REACTOR

MULTI-APPLICATION SMALL LIGHT WATER REACTOR

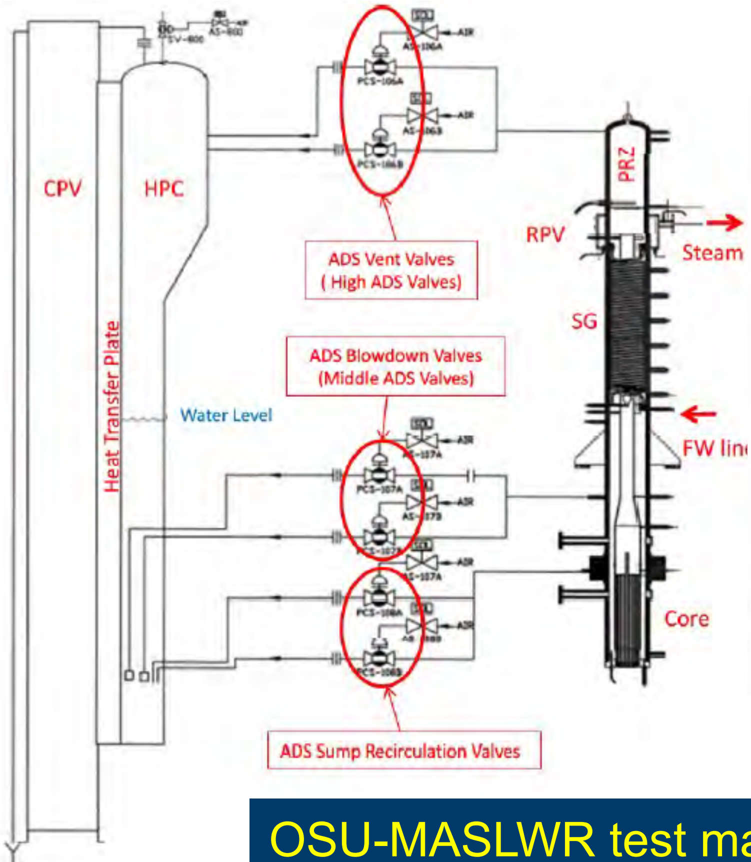
- The Multi-Application Small Light Water Reactor (MASLWR) project was conducted under the auspices of the Nuclear Energy Research Initiative (NERI) of the U.S. Department of Energy (DOE).
- The primary project objectives were to develop the conceptual design for a safe and economic small, natural circulation light water reactor,
    - to address the economic and safety attributes of the concept,
    - to demonstrate the technical feasibility by testing in an integral test facility.
- The testing program has been conducted at Oregon State University (OSU).
- MASLWR is the basis for the NuScale Power Integral Reactor Design.

ENEA

MULTI-APPLICATION SMALL LIGHT WATER REACTOR



OREGON STATE UNIVERSITY (OSU) MASLWR FACILITY



OSU-MASLWR Main Scaling Parameters

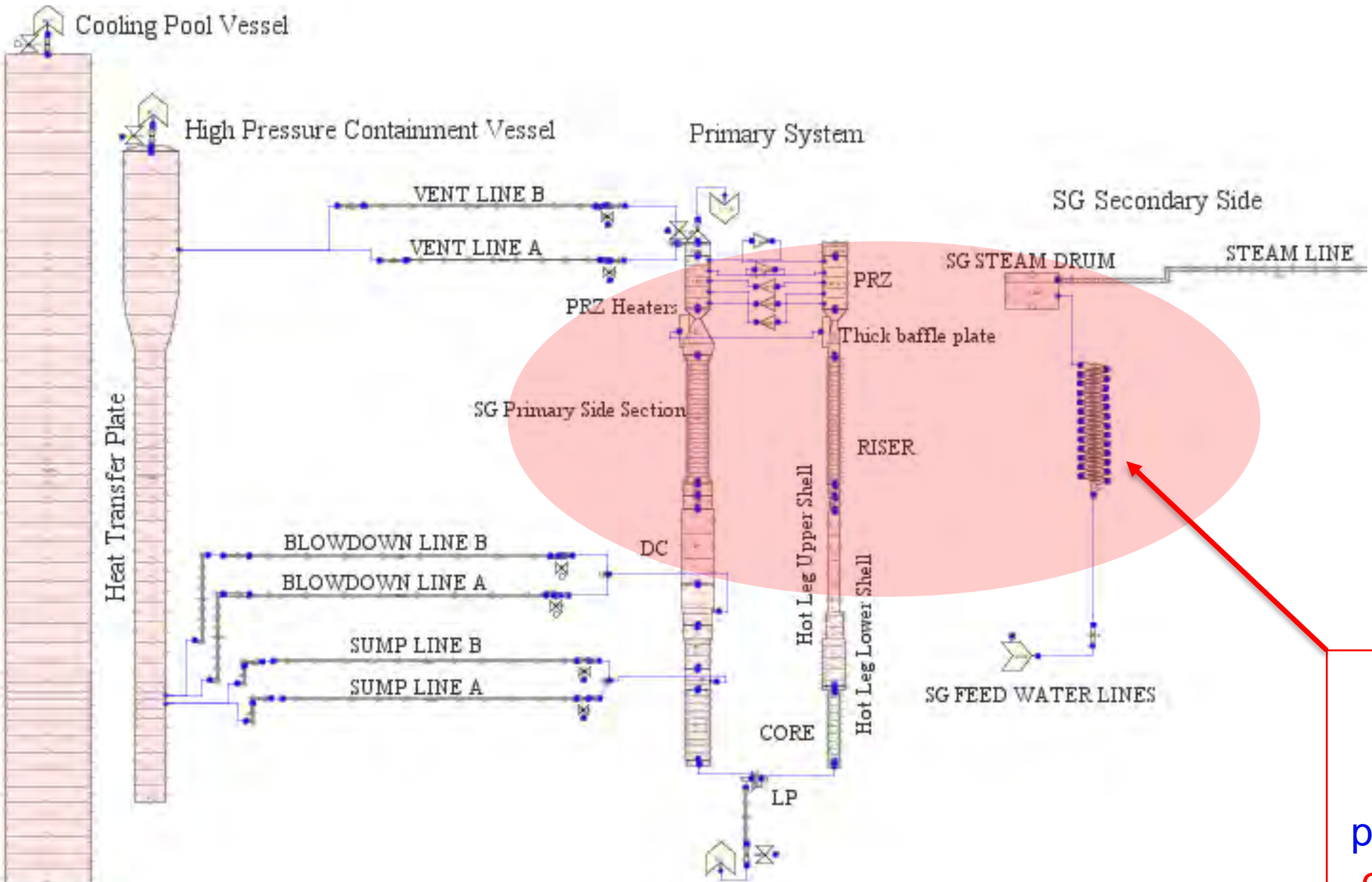
Parameter	MASLWR	OSU-MASLWR	Note
Time Scale	1:1	1:1	Time preserved facility
Volumetric Scale	1:1	1:254.7	Reduced volume facility
Height Scale	1:1	1:3	Reduced height facility
Design Primary Pressure (MPa)	8.6	11.4	Full pressure facility
Loop Number	1	1	Integral test facility
Hot Leg Riser D (mm)	914.4	102.3	
Power (MW)	150	0.6	Full power facility
Core Rod Number	6336	56	
Core Rod DIA (mm)	9.5222	15.9	
Heated Length (m)	1.35	0.686	
SG Type	Vertical once-through helical tubes	Vertical once-through helical tubes	
SG Bundle	2	3	
SG Tubes Number For Bundle	506	5, 5, 4	
SG OD Tube (mm)	15.9	15.9	

OSU-MASLWR test matrix available at the International Community

	Test	Test Type
DOE	OSU-MASLWR-001	Inadvertent Actuation of 1 Submerged ADS Valve
	OSU-MASLWR-002	Natural Circulation at Core Power up to 210 kW
	OSU-MASLWR-003A	Natural Circulation at Core Power of 210 kW
	OSU-MASLWR-003B	Inadvertent Actuation of 1 High Containment ADS Valve
IAEA	ICSP test SP-2	Loss of Feedwater Transient
	ICSP test SP-3	Power Maneuvering

OSU-MASLWR TRACE MODEL

- The analyses have been conducted with the TRACE (TRAC/RELAP Advanced Computational Engine) best-estimate thermal-hydraulic system code developed by USNRC.



Nodalization Strategy

Slice Nodalization

Realize the same dimension in nodes of different zones of the nodalization simulating zones of the plant at the same elevation

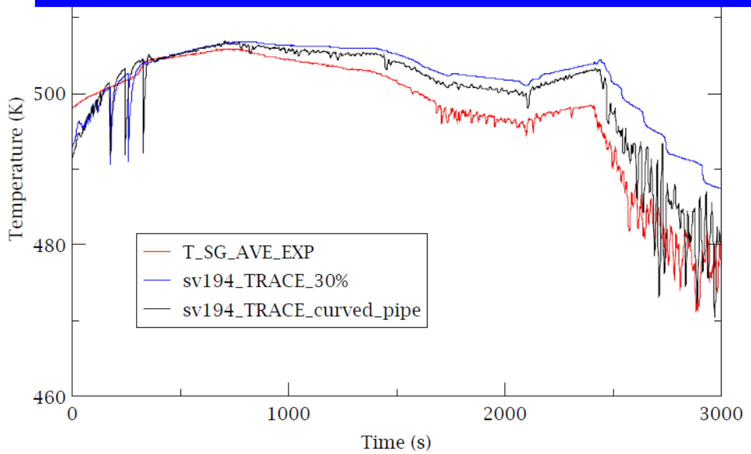
Improve Natural Circulation Performance in the code

The SG coils are simulated with one equivalent group of pipes. The curved pipe option has been used

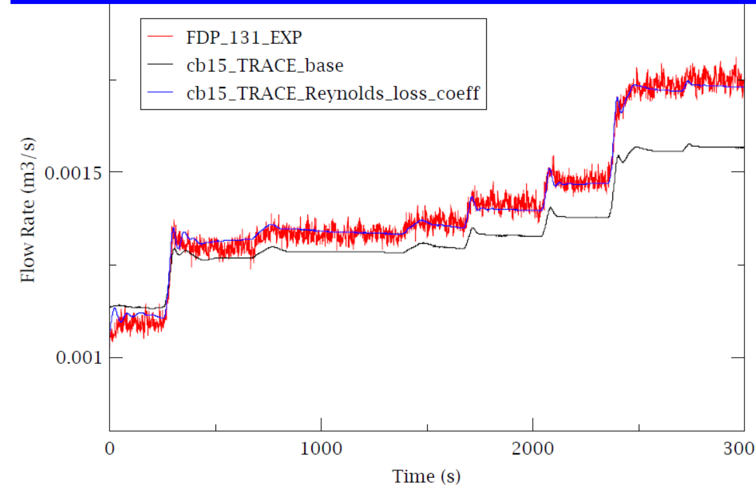
OSU-MASLWR NATURAL CIRCULATION TESTS: TRACE PREDICTION VS EXPERIMENTAL RESULTS

OSU-MASLWR-002 test

OSU-MASLWR-002, experimental data versus code calculations for the average value of the SG outlet temperature

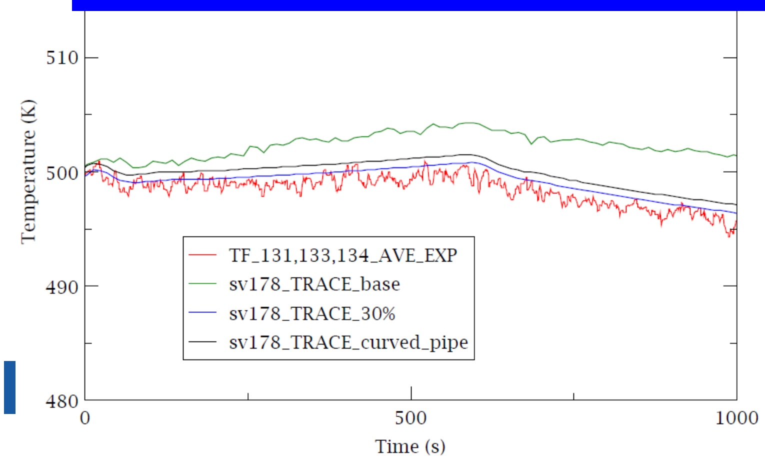


OSU-MASLWR-002, experimental data versus code calculations for the primary volumetric flow rate

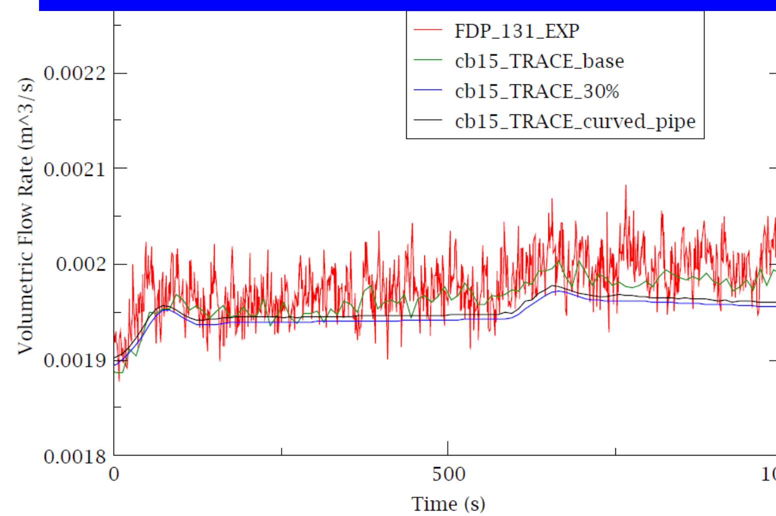


OSU-MASLWR-003A test

OSU-MASLWR-003A, experimental data versus code calculations for the average value of the SG outlet temperature

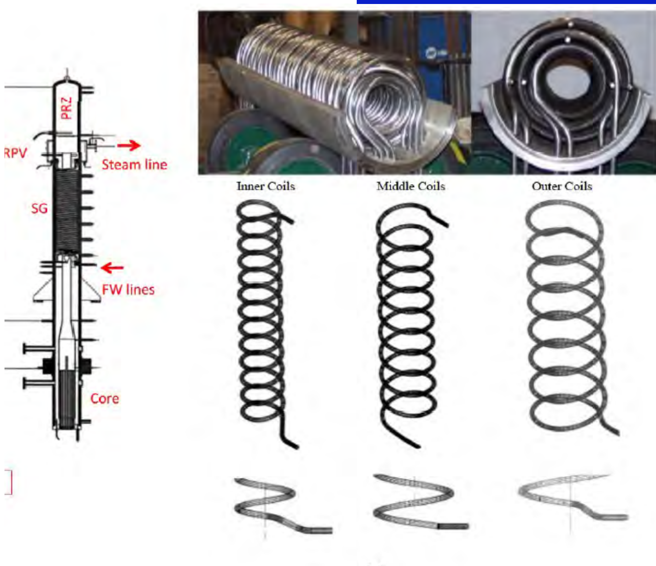


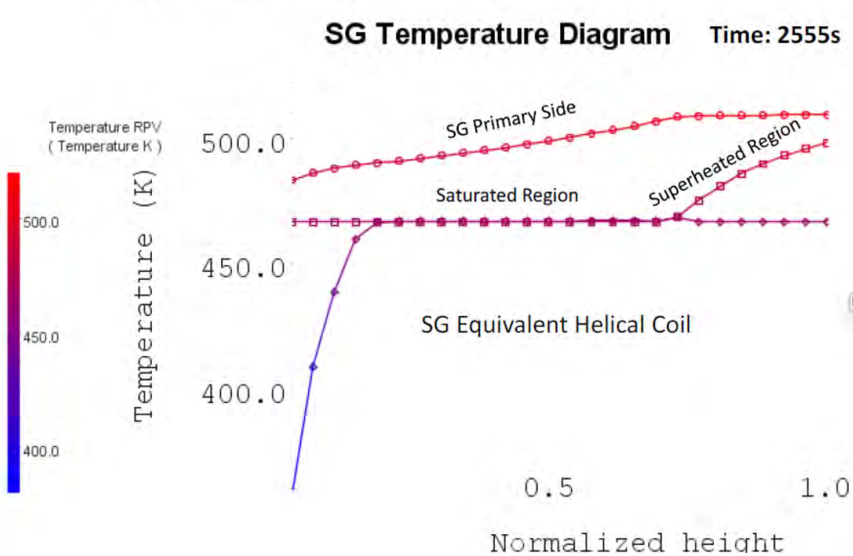
OSU-MASLWR-003A, experimental data versus code calculations for the primary volumetric flow rate

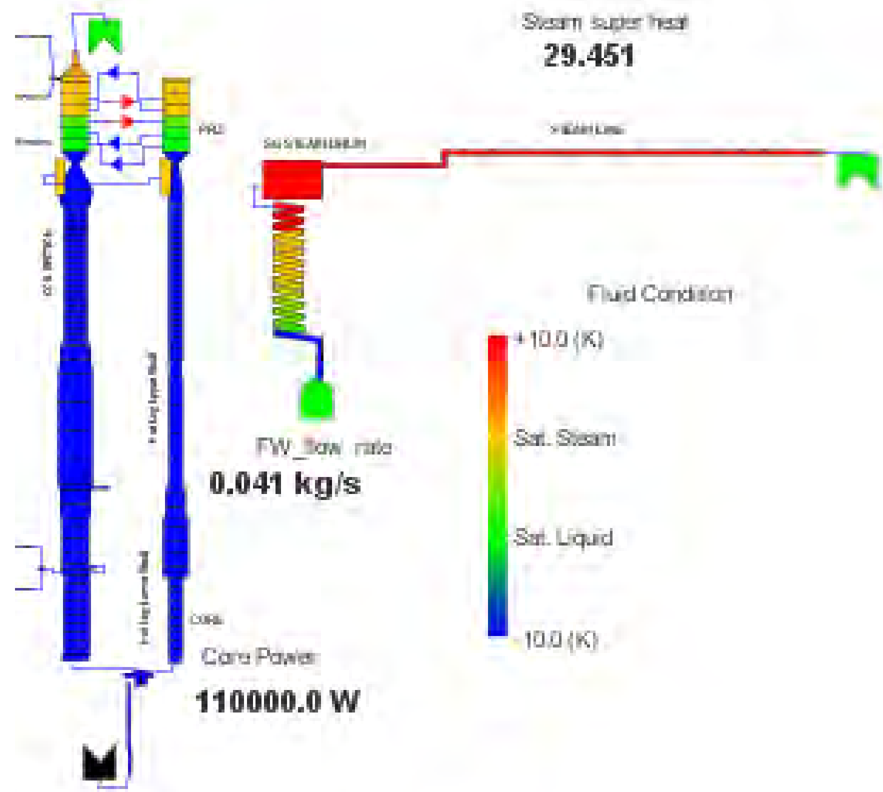


OSU-MASLWR NATURAL CIRCULATION TESTS: TEST VERSUS PHENOMENA, TRACE PREDICTION VS PHENOMENA

OSU-MASLWR and Helical Coil Compact SG







OSU-MASLWR NATURAL CIRCULATION TESTS: TEST VERSUS PHENOMENA, TRACE PREDICTION VS PHENOMENA

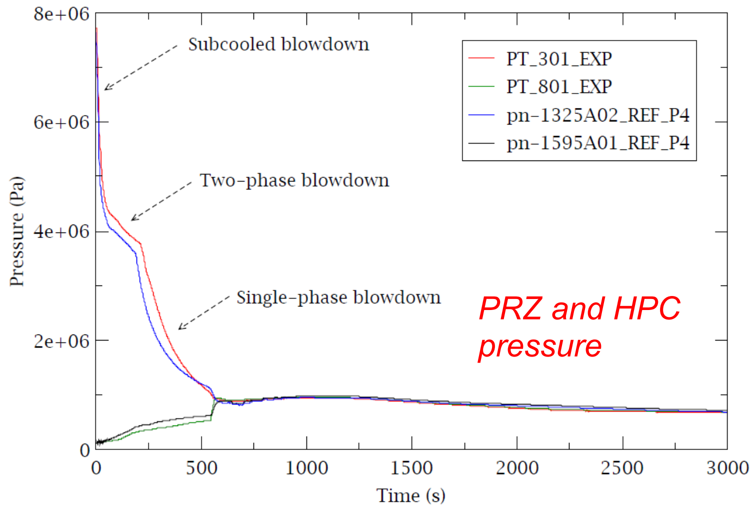
Phenomenon	Experiment		TRACE code
	Phenomena	Measurement	Phenomena
Single-phase Natural Circulation	+	+	+
Heat Transfer in Covered Core	+	+	+
By Pass Heat Transfer	+	+	+
Distribution of Pressure Drop Through Primary System	+	+	+
Heat Transfer in SG Primary Side	+	+	+
Structural Heat and Heat Losses	+	o	+
Heat Transfer in SG Secondary Side	+	+	+*
Steam Superheated on Secondary Side	+	+	+*

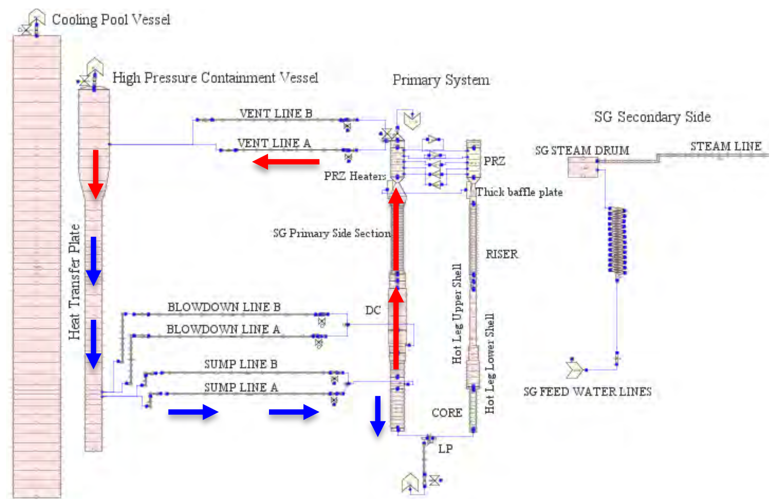
**The curved pipe option has been used for simulating flow through a curved pipe such as secondary side of a helical steam generator. This option is available from TRACE V5 patch 4. In general the model is able to quantitatively better predict heat transfer between primary and secondary side through an helical coil SG. However still some quantitative difference are present. The quantitative analyses done by FFTBM for the OSU-MASLWR-002 and 003A test show that the quantitative discrepancies are higher when the FW mass flow rate increase.*

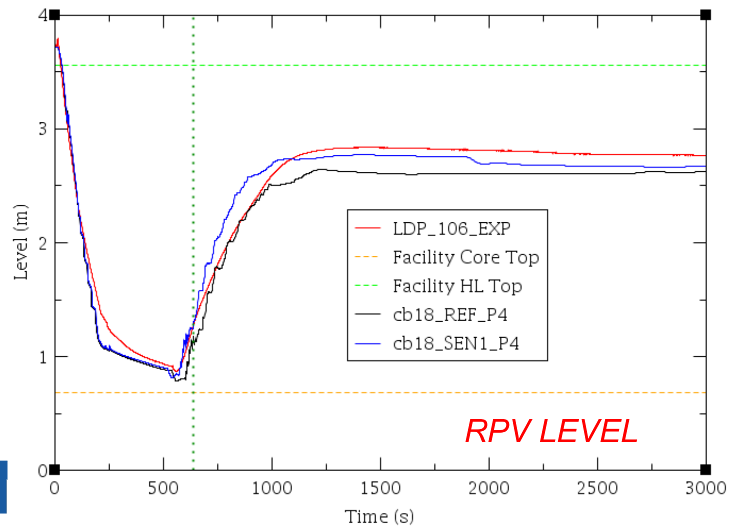
	+	o	NA	-
Experimental data	Phenomenon occurred in the test and it is directly measured	Phenomenon occurred in the test and it is indirectly measured	Phenomenon occurred during the test but there is no instrumentation to detect (lack of instrumentation)	Phenomenon not occurred in the test
Calculated data	Phenomenon is clearly predicted by the code **(Excellent/Reasonable)**	Phenomenon is partially predicted (i.e. the answer of the code is **reasonable** but closure code relation are not appropriate, etc)	Models are not appropriate to predict (i.e. nodalization strategy, etc) **(Minimal)**	Phenomenon is not predicted by the code **(Unqualified)**

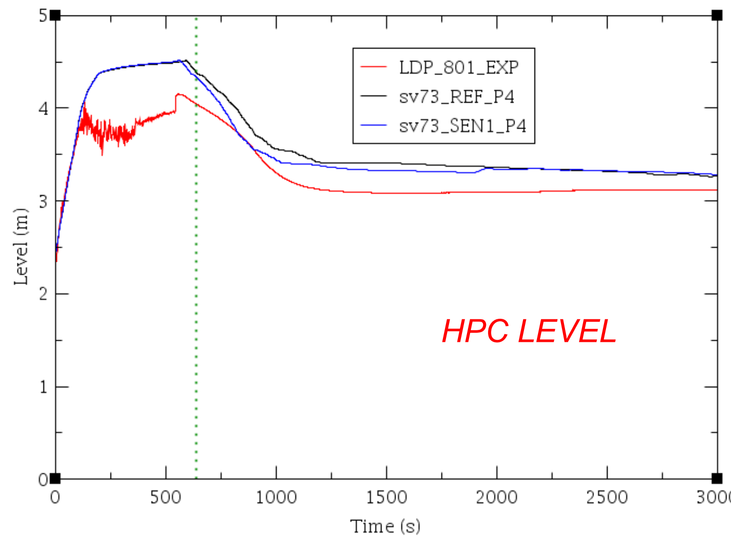
OSU-MASLWR PASSIVE PRIMARY/CONTAINMENT NATURAL CIRCULATION MITIGATION STRATEGY TESTS: TRACE PREDICTION VS EXPERIMENTAL RESULTS

OSU-MASLWR-001, experimental data versus code calculations





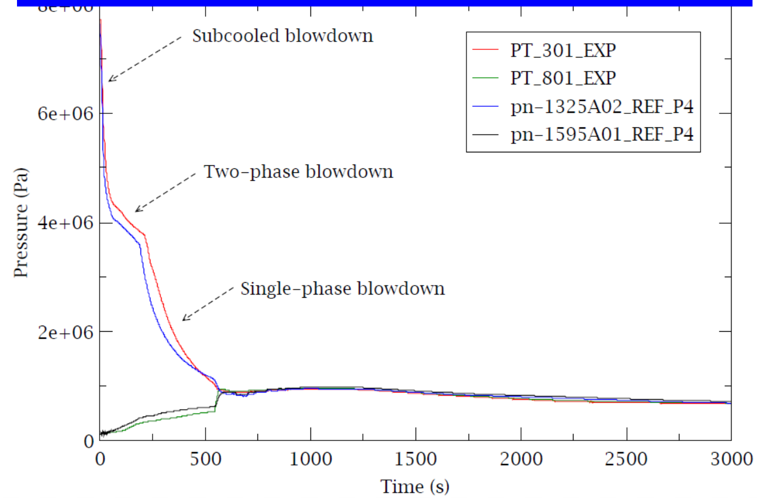




OSU-MASLWR PASSIVE PRIMARY/CONTAINMENT NATURAL CIRCULATION MITIGATION STRATEGY TESTS: TRACE PREDICTION VS EXPERIMENTAL RESULTS

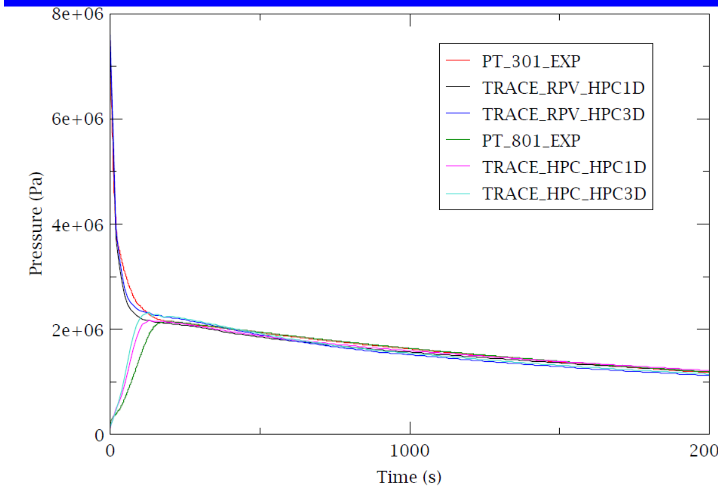
OSU-MASLWR-001 test

OSU-MASLWR-001, experimental data versus code calculations for the PRZ and HPC pressure



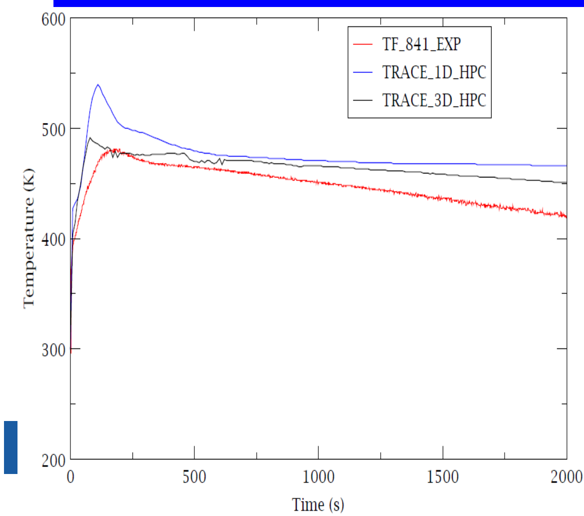
OSU-MASLWR-003B test

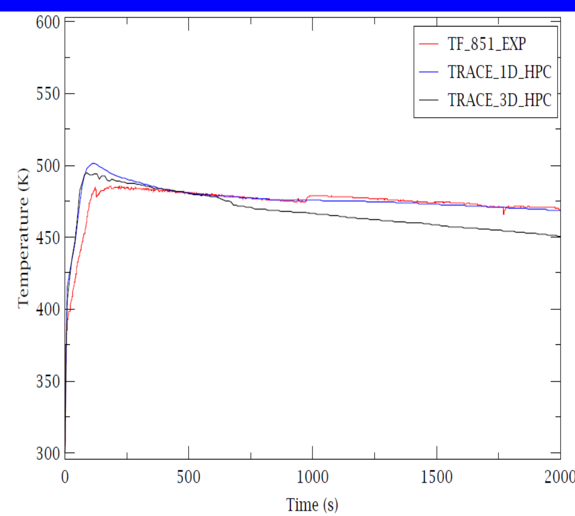
OSU-MASLWR-003B, experimental data vs code calculation for the primary and HPC pressure

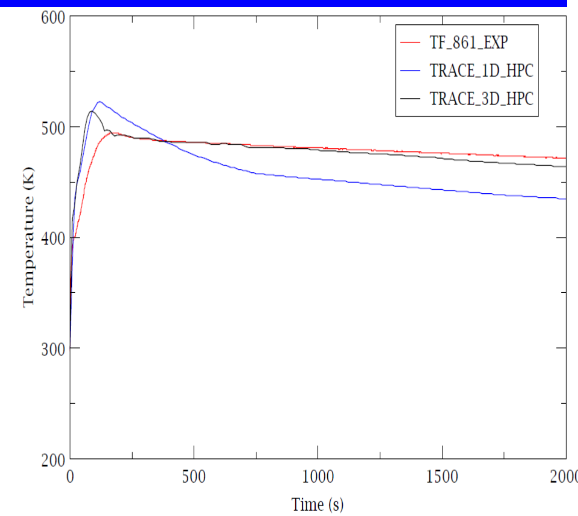


OSU-MASLWR-003B test

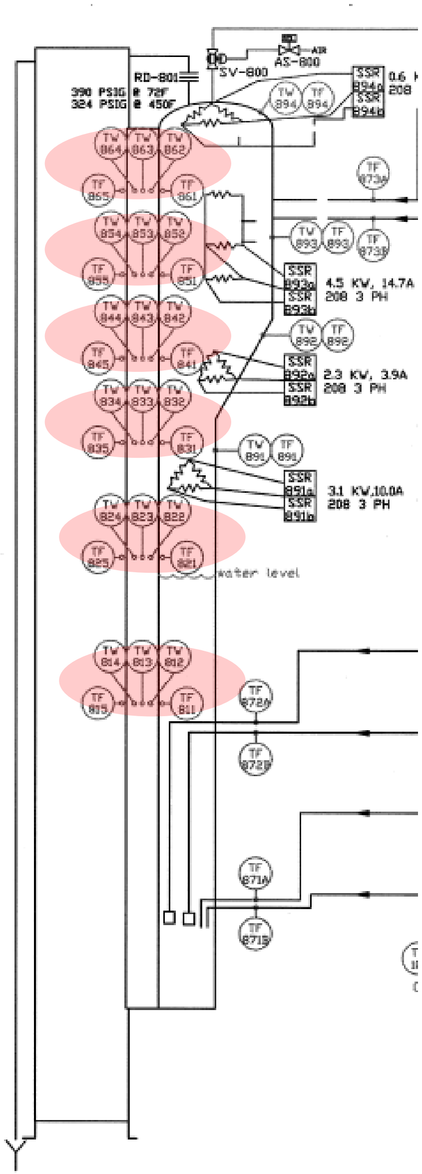
OSU-MASLWR-003B, experimental data versus code calculations for some HPC temperatures

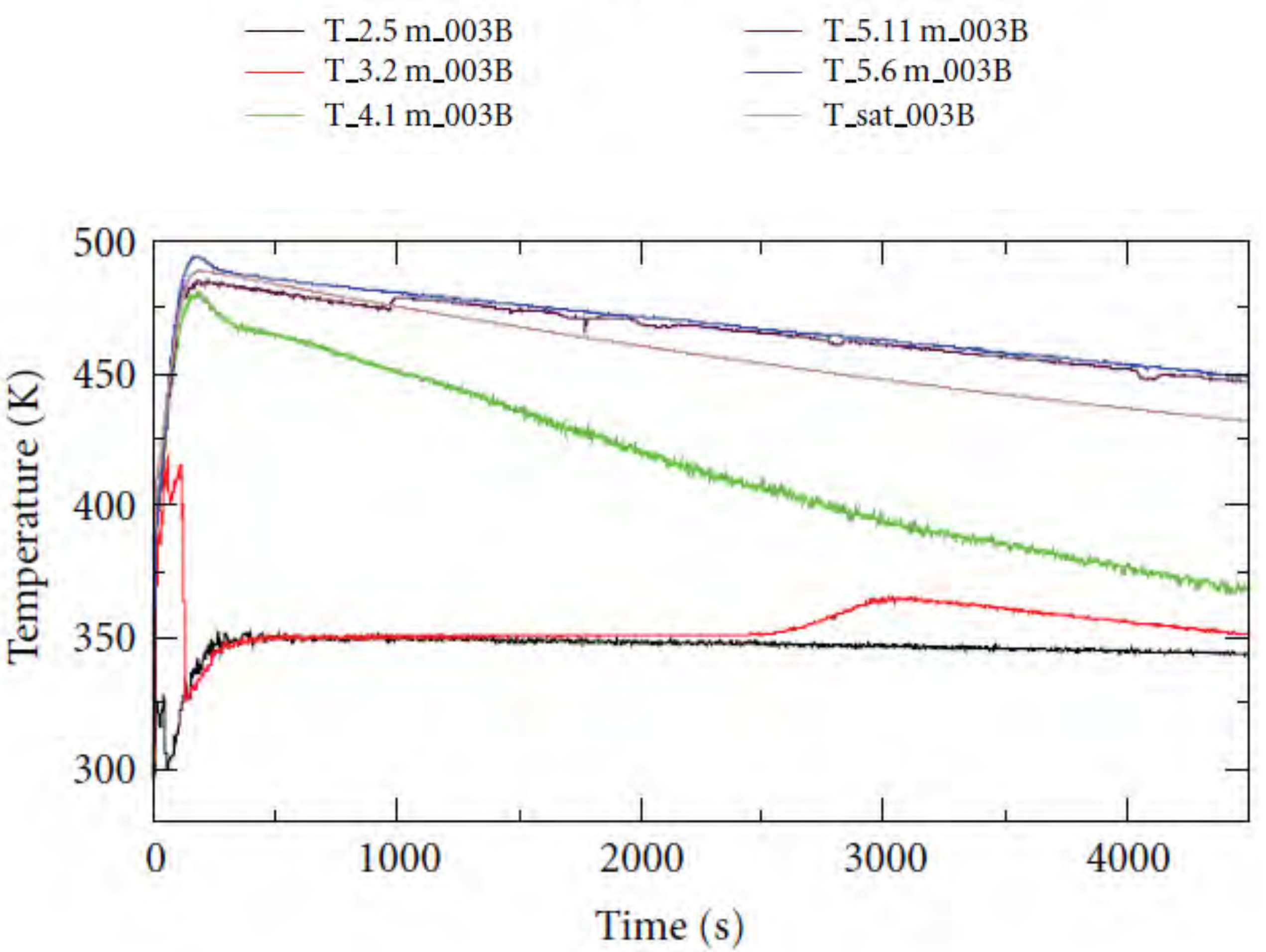




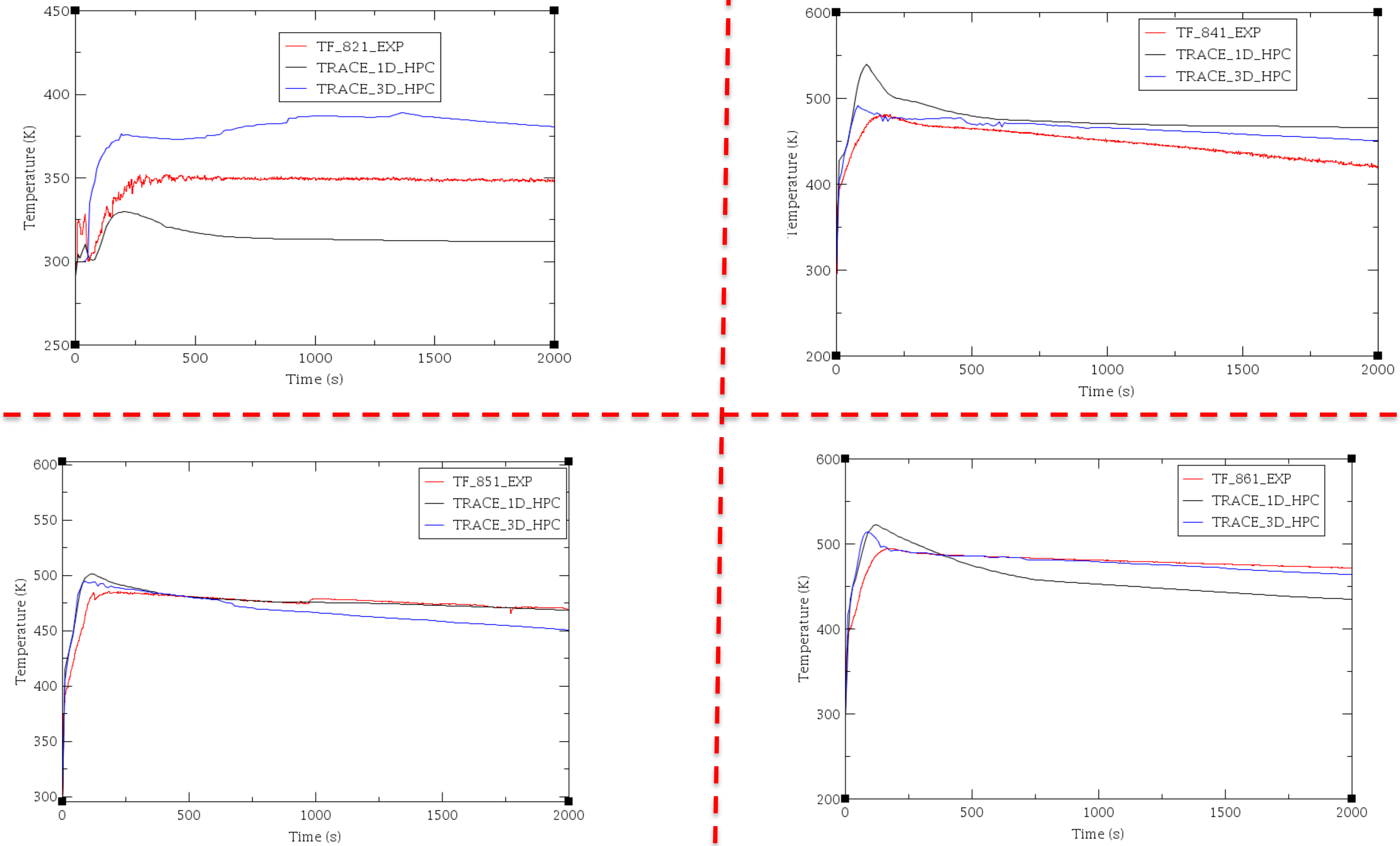


POST TEST ANALYSIS OF THE OSU-MASLWR 003B TEST-HPC TEMPERATURE





POST TEST ANALYSIS OF THE OSU-MASLWR 003B TEST-HPC TEMPERATURE



OSU-MASLWR PASSIVE PRIMARY/CONTAINMENT NATURAL CIRCULATION MITIGATION STRATEGY TESTS: TEST VERSUS PHENOMENA, TRACE PREDICTION VS PHENOMENA

Phenomenon	Experiment		TRACE code
	Phenomena	Measurement	Phenomena
Single-Phase Natural Circulation	+	+	+
Two-Phase Natural Circulation	+	NA	+
Distribution of Pressure Drop Through Primary System	+	+	+
Primary-Containment Coupling During Blowdown And Long Term Cooling	+	o	+
Structural Heat And Heat Losses	+	o	+
Break Flow	+	o	+
Behavior of Large Pool: Thermal Stratification (HPC)	+	+	+
Behavior of Large Pool: Natural Convection (HPC)	+	NA	NA (1D)* + (3D) *
Behavior of Large Pool: Steam Condensation (HPC)	+	NA	+
Effect of Non-Condensable Gases On Condensation Heat Transfer (HPC)	+	NA	+
Condensation on Containment Structures (HPC)	+	o	+
Behavior of Large Pool: Thermal Stratification (CPV)	+	+	+
Behavior of Large Pool: Natural Convection (CPV)	+	NA	NA*

**The natural circulation phenomena in HPC and CPV are not predicted by the TRACE code for the 1D nodalization strategy of the HPC and CPV. A 3D model, by using the vessel component, permits the prediction of these phenomena in the HPC (this option has been tested for the OSU-MASLWR 003B).*

MODELING LESSONS LEARNED

- The analysis of the natural circulation tests (OSU-MASLWR-002 and 003A) shows that TRACE is able to qualitatively and quantitatively predict natural circulation phenomena and heat transfer from primary to secondary side by helical coil SG. In particular it is suggested the adoption of
  - the “curved pipe” option available in TRACE code to properly simulate the heat transfer in helical coiled SG. If specific models are not available, the heat transfer coefficient should be increased (around 30%)
  - the Reynolds number dependent form losses to correctly predict the different primary flow rate values.

- The analysis of the blowdown tests (OSU-MASLWR-001 and 003B) shows that TRACE is able to qualitatively and quantitatively predict the single and two-phase natural circulation and primary/containment coupling phenomena. A more detailed 3D HPC model, by using the TRACE “3D vessel” component, may provide a better quantitative estimation of the containment temperatures.

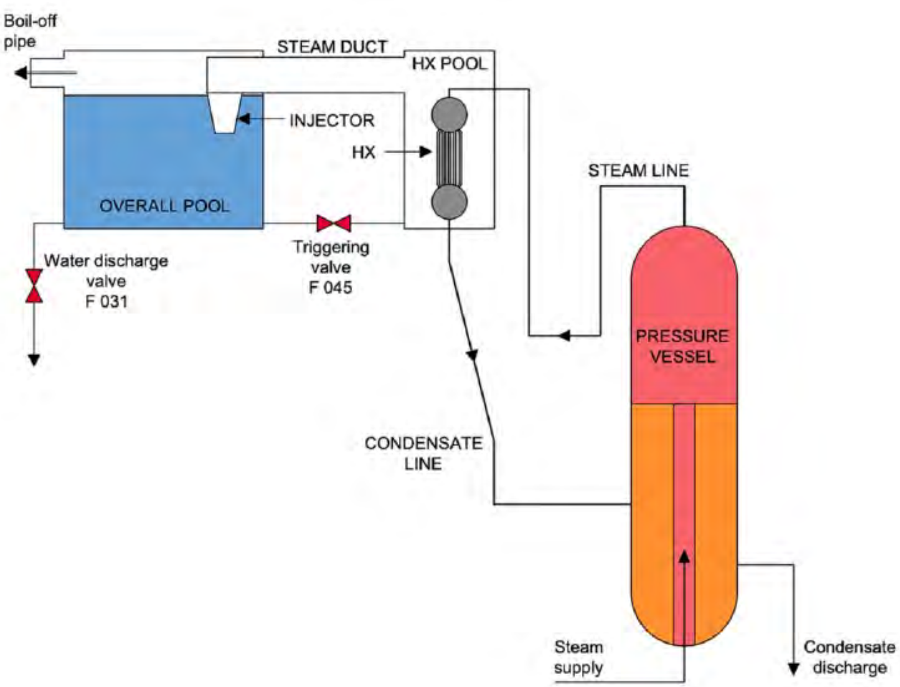
PERSEO FACILITY

PERSEO FACILITY

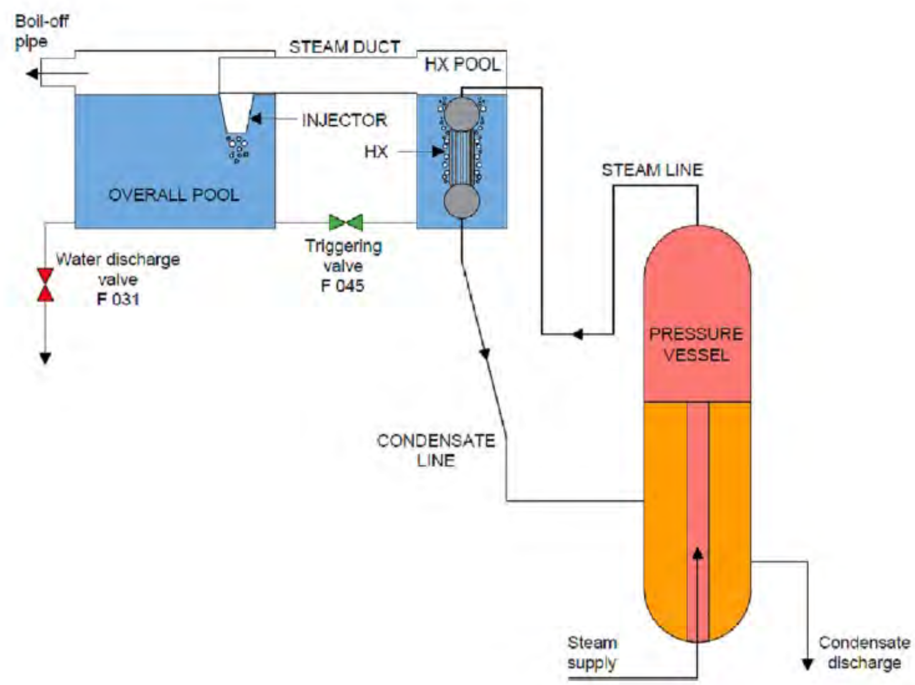
- PERSEO is a full-scale SETF-component test aimed at studying a new passive decay heat removal system operating in natural circulation;
- It was built at SIET (Italy) modifying the existing PANTHERS IC-PCC facility;
- PERSEO was not designed to simulate a specific passive system of a particular reactor design currently in operation or under design, but its main purpose is the assessment of the performance and the efficiency of a new in-pool heat exchanger for decay heat removal, implementing natural circulation.
- The facility is characterized by full height, full volume, prototypical fluid conditions and a maximum power of around 20 MW.
- It is a full pressure facility, in fact can operate up to conditions typical of a BWR or the secondary side of a PWR.
- An international open benchmark was conducted in the framework of the OECD/NEA/CSNI activity on the “Status report on thermal–hydraulic passive systems design and safety assessment”.

PERSEO FACILITY

Passive system not in operation



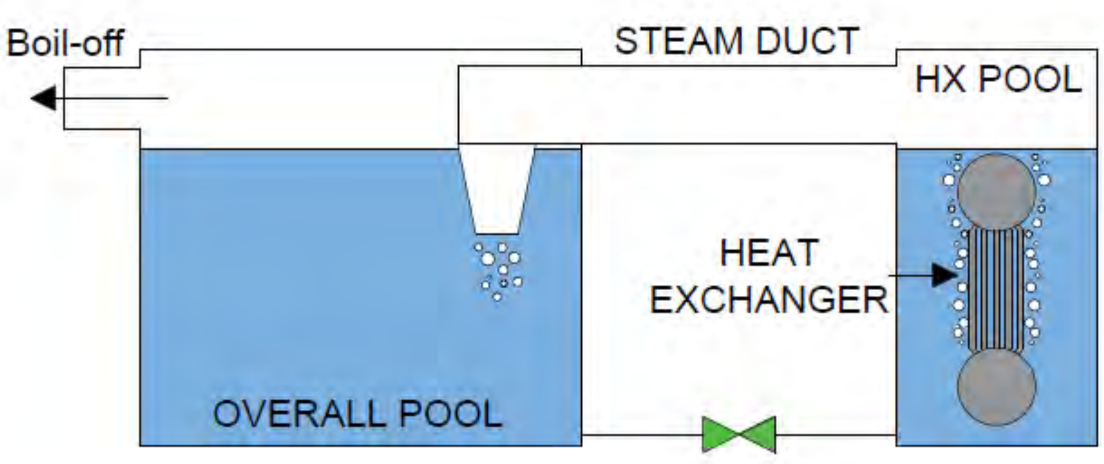
Passive system in operation



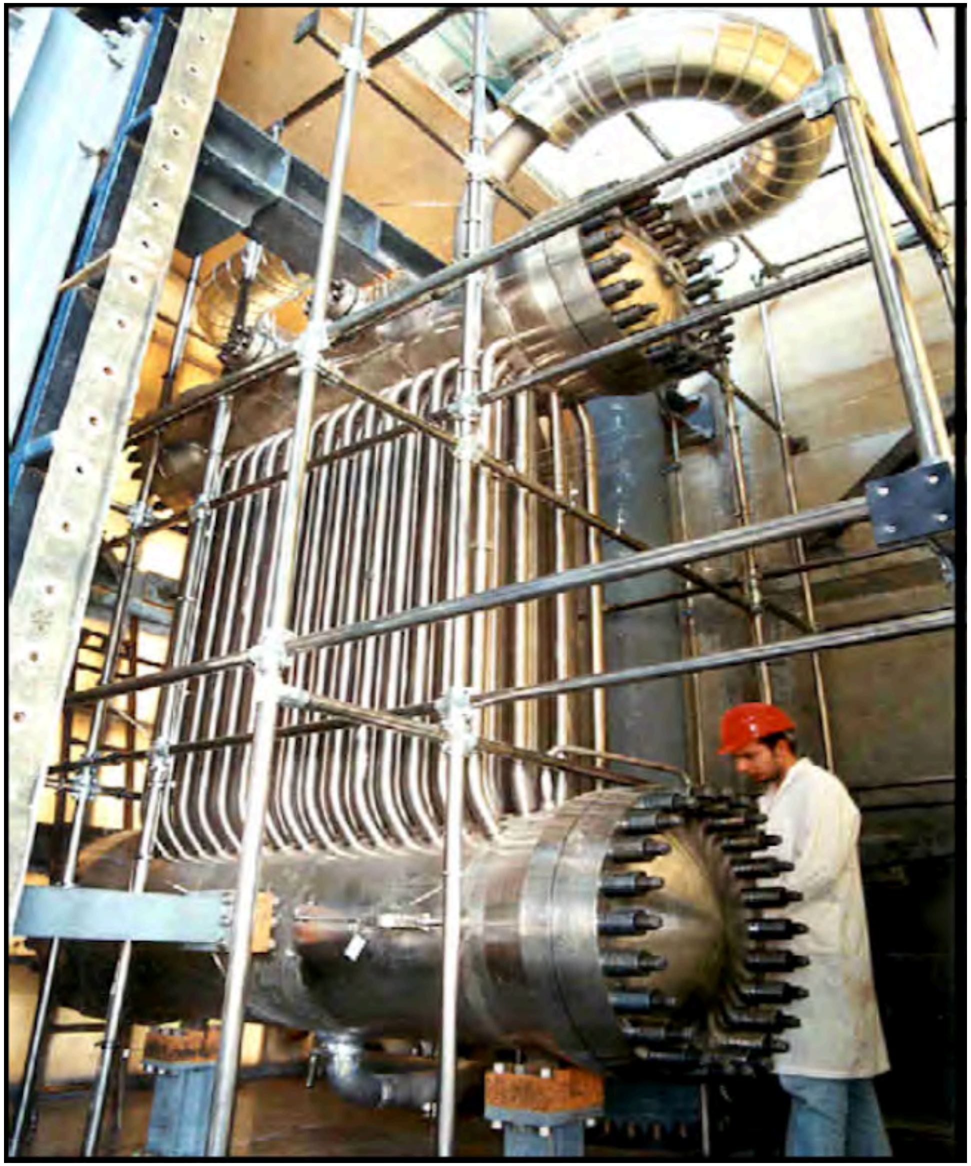
ENEA

PERSEO FACILITY

- The data collected in PERSEO facility are particularly useful for the analysis of the behavior of heat sinks of passive systems:
  - In-tube condensation;
  - Pool-side nucleate boiling;
  - Pool-side thermal stratification;
  - Coupled pools dynamics.
- Two tests have been analyzed and simulated:
  - Test 7 at around 70 bar;
  - Test 9 at around 40 bar.



PERSEO FACILITY



PERSEO BENCHMARK

- In the framework of the OECD/NEA/CSNI/WGAMA group, an activity on the "Status report on thermal-hydraulic passive systems design and safety assessment" has been conducted. University of Pisa was the lead Organization.

- Computational tools adopted to perform safety analysis for design and licensing of nuclear plants should be tested for passive systems.

- A benchmark exercise, based on the experimental data developed in the PERSEO (in-Pool Energy Removal System for Emergency Operation) facility has been proposed by ENEA and approved by the "WGAMA task group on thermal-hydraulic of passive system”.

- Computational tools have been tested for passive systems using PERSEO data.

- ENEA organized and led all the benchmark activities, including the official distribution of experimental data.

BENCHMARK PARTICIPANTS

COUNTRY	INSTITUTION	CODE	RESULTS AND REPORT DELIVERED
Canada	CNSC/CCSN	-	Cancelled
China	SPICRI	-	Cancelled
Croatia	FER	-	Cancelled
Czech Republic	UJV	RELAP5/MOD3.3Patch05	✓
		MELCOR v2.	Cancelled
		GOTHIC	Cancelled
Finland	VTT	Apros 6.09	✓
France	IRSN	CATHARE	Cancelled
Germany	GRS	ATHLET 3.2	✓
Italy	POLITO	RELAP5-3D v4.3.4	✓
	UNIPI	RELAP5/MOD3.3Patch03	✓
	UNIROMA1	RELAP5-3D	✓
Netherland	NRG	SPECTRA v3.61	✓
Russia	IBRAE	SOCRAT	✓
South Korea	KAERI	MARS-KS v1.4	✓
		SPACE v3.2	✓
	KINS	-	Cancelled
Spain	UPV	-	Cancelled
USA	NuScale	-	Cancelled
	UofU	MELCOR v2.2.11620	✓
	OSU	-	Cancelled
	USNRC	TRACE V5.1175	✓

- 19 Organizations worldwide expressed their interest to participate in the PERSEO benchmark.
- 12 results from 11 Organizations were submitted.
- Noticeably, 7 out of 12 calculations have been carried out by code developers.
- Most of the safety analysis computational tools adopted by the international community have been applied.
- The benchmark activity lasted about 2 years (2018-2020).

PERSEO BENCHMARK- MODELS IMPLEMENTED IN THE CODES FOR THE HX HEAT TRANSFER

CODE	Pool-side heat transfer	Tube-side heat transfer
Apros 6.09	Convection: Dittus-Boelter Boiling: Thom correlation	Chen (with Vierow-Schrock correction for non-condensable gases)
ATHLET 3.2	Modified model of Chen (for subcooled and saturated nucleate boiling) and the model of Mc Adams (natural convection)	The Models of Chen (macroscopic part only, forced convection), Nusselt (laminar condensation) and Carpenter and Colburn (turbulent condensation) have been replaced by models of Nusselt (laminar condensation), Kutateladze (laminar wavy condensation) and Chen (turbulent condensation)
MARS-KS v1.4	Chen model	Maximum value of Shah and Nusselt models
MELCOR v2.2.11620	Convection, Nucleate boiling (Chen), CHF (Zuber), minimum film boiling heat flux (Zuber), film boiling (Bromley)	MELCOR calculates a diffusion mass transfer coefficient using the Sherwood number correlation to model condensation. Condensed materials are then computed by MELCORs liquid film model
RELAP5-3D v4.3.4.	Nucleate boiling (Chen), transition boiling (Chen), and film boiling (Bromley), natural convection (Churchill-Chu or McAdams)	Laminar condensation (Nusselt), turbulent condensation (Shah), presence of non-condensable gases (Coburn-Hougen)
RELAP5/MOD3.3 Patch05	Nucleate (Chen-Inayatov), transition (Chen), and film (Bromley) modes are dominant	Nusselt-Shah-Coburn-Hougen
RELAP5/MOD3.3 Patch03	Nucleate (Chen-Inayatov), transition (Chen), and film (Bromley) modes are dominant	Nusselt-Shah-Coburn-Hougen

SOCRAT V1	Chen (for boiling) Dittus (convection)	Combined correlations from Dittus, Bromley&Rohsenow and coeff * Ra^m * Pr^n
SPACE v3.2	Chen model	Maximum value of Shah and Nusselt models
SPECTRA v3.61	Chen (nucleate boiling), McAdams (natural convection)	Nusselt correlation for laminar flow in vertical tubes, with multipliers to account for the presence of non-condensables and shear stress in turbulent regime
TRACE V5.1175	Single phase: Pipe: Laminar forced convection – Nusselt analytical formula Turbulent forced convection – Gnielinski Natural convection - traditional model with function of Raleigh number Rod bundle – El-Genk Boiling Nucleate boiling (Chen formulation with Gorenflo correlation for pool boiling component) Transition boiling - Interpolation between CHF and Minimum Film Boiling (MFB) Film boiling (Inverted annular – Cachard; Dispersed flow: radiative flux to both phase plus convective flux to superheated steam)	Kuhn-Schrock-Peterson (K-S-P) for laminar film condensation, Modified Gnielinski Correlation for turbulent film condensation. Noncondensable effect -Coburn-Hougen

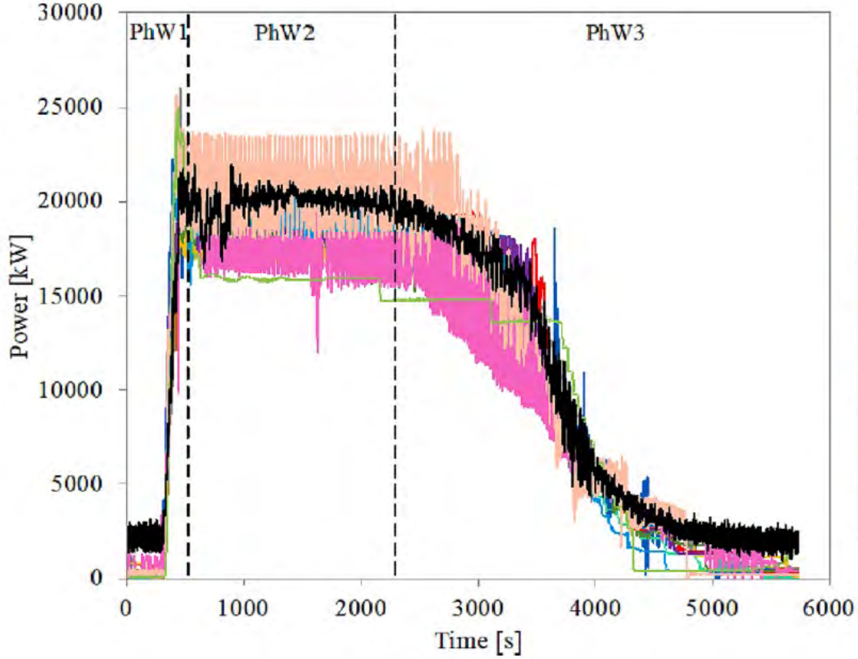
USER APPROACH FOR THE SIMULATION OF THE HEAT TRANSFER IN THE HEAT EXCHANGER

Participant	Code	Note
GRS	ATHLET 3.2	MODIFIED VERSION OF ATHLET 3.2: For the tube-side, the models of Chen (macroscopic part only, forced convection), Nusselt (laminar condensation) and Carpenter and Colburn (turbulent condensation) has been replaced by the models of Nusselt (laminar condensation), Kutateladze (laminar wavy condensation) and Chen (turbulent condensation) (Papini and Cammi, 2010)
IBRAE	SOCRAT V1	NO USER TUNING OR NEW CORRELATIONS IMPLEMENTED
KAERI_a	MARS-KS v1.4	USER TUNING: Heat transfer coefficient in the HX was multiplied by 2.4
KAERI_b	SPACE v3.2	USER TUNING: Heat transfer coefficients for the nucleate boiling and condensation was multiplied by 3.0.
NRG	SPECTRA v3.61	USER TUNING: In order to improve the prediction of the power removal, the value of the characteristic length for the film condensation adopted is 5 cm, corresponding to a multiplier of 2.44 for the HTC. Additionally, the opening fraction of the triggering valve was tuned in order to obtain the expected mass flow rate.
POLITO	RELAP5-3D v4.3.4.	USER TUNING: To correctly reproduce the HX power a multiplication factor for the heat transfer coefficients has been applied on the inner and outer side of the HX heat structure. This multiplication factor, called fouling factor in RELAP5-3D, has been set equal to 2.4

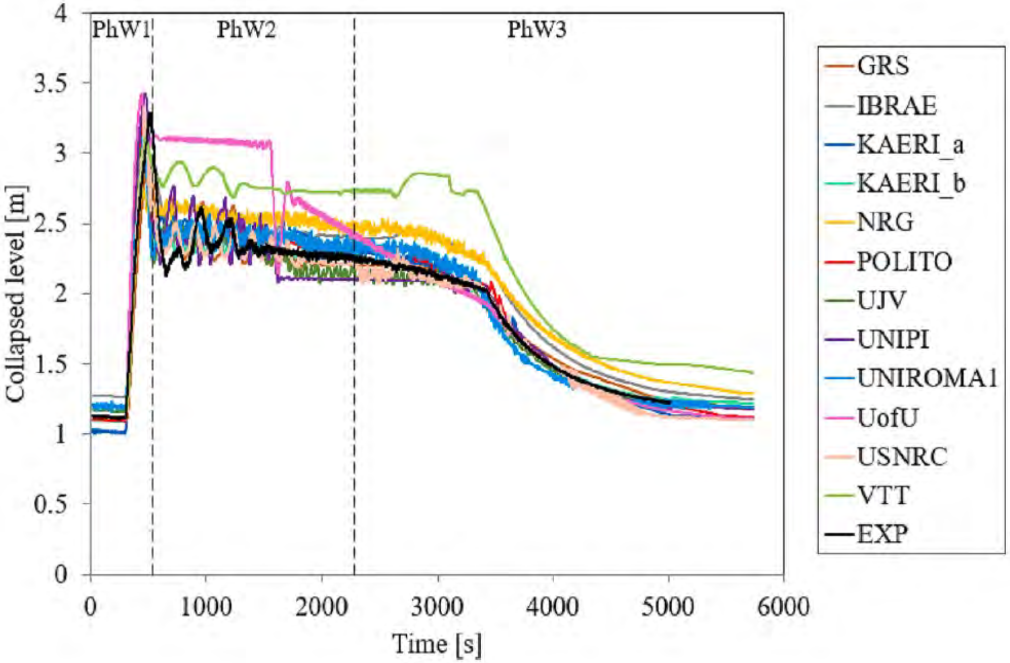
Participant	Code	Note
UJV	RELAP5/ MOD3.3 Patch05	USER TUNING: The HX power underestimation was compensated by means of a fouling factor (multiplier for heat transfer coefficients) value of 2.2 applied on the heat transfer coefficients on both sides of HX tubes.
UNIPI	RELAP5/ MOD3.3 Patch03	USER TUNING: Tuning is the application of a multiplication factor for the heat transfer coefficients on the inner and outer side of the HX heat structure. This multiplication factor, called fouling factor in RELAP5 has been set equal to 2.4
UNIROMA1	RELAP5-3D v4.3.4.	USER TUNING: A multiplicative factor equal to 2.4 has been introduced in both HX sides.
UofU	MELCOR v2.2.11620	USER TUNING: Several sensitivity coefficients in the Heat Structure Package are modified. Initial versions of the MELCOR model of the PERSEO facility used a multiplier of 120 to match the PERSEO facility description supplied, however, this value was increased to 180 in the final version of the model.
USNRC	TRACE V5.1175	NO USER TUNING OR NEW CORRELATIONS IMPLEMENTED
VTT	Apros 6.09	NO USER TUNING FOR THE HEAT TRANSFER

PERSEO TEST 7 PART 2 RESULTS

HX POWER



HXP WATER LEVEL



MODELING LESSONS LEARNED

- Based on analyses developed along the OECD/NEA/CSNI/WGAMA PERSEO benchmark the majority of the codes showed an underestimation of the heat transfer, which was adjusted with different approaches to better predict the experimental transient behavior.
- The PERSEO benchmark showed the limitations of most of the codes in predicting the full scale experimental results without modifications of the code models or of the nodalization, which in this case were possible thanks to the "open" nature of the benchmark, with the experimental data provided to the Participants before the beginning of the calculation phase.
- Therefore, an accurate evaluation of the validation domain and of the related scaling issues is fundamental for the application of the codes in the simulation of full scale passive systems.
- the simulation of large pools, 1D nodalization, fictitious 2D nodalization and 3D components, available in some codes, have been tested. However, additional activities would be needed to evaluate which approach may provide the most accurate results (e.g. concerning the thermal stratification) with respect to the experimental data.

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SCALING

Scaling analysis

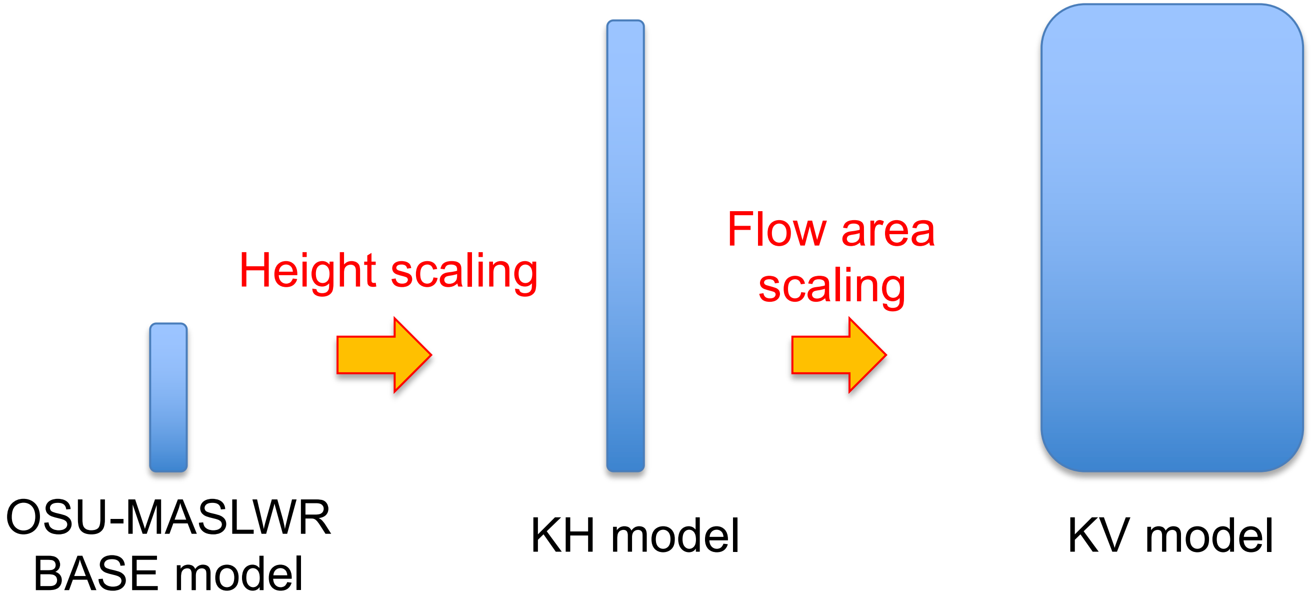
- *OSU-MASLWR natural circulation tests (TRACE code)*

SCALE-UP METHODOLOGY

- The OSU-MASLWR facility is called in height (1:3) and in volume (1:254.7)
- The scale-up of the TRACE input-deck has been done in two subsequent steps:

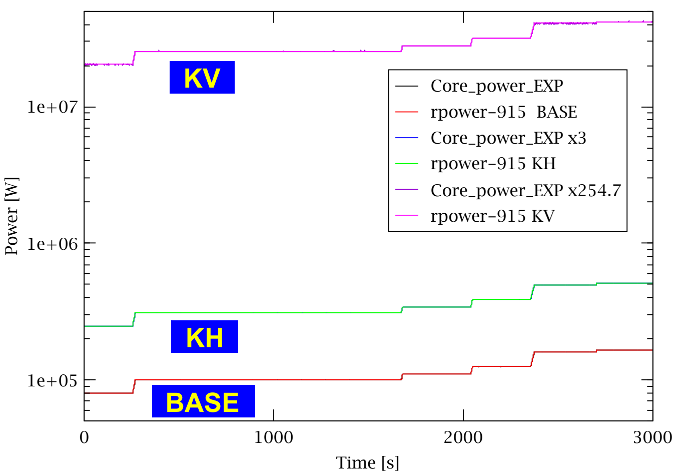
  1) Height scaling → 2) Flow area scaling

- The evaluation of the results are based on the comparison of some relevant characteristic and non-dimensional numbers and properly scaled thermal-hydraulic parameters.

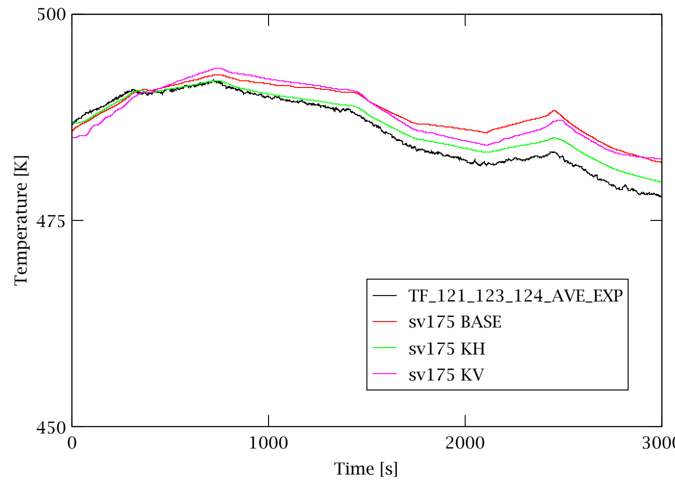


COMPARISON OF THE CALCULATIONS (BASE, KH, KV)

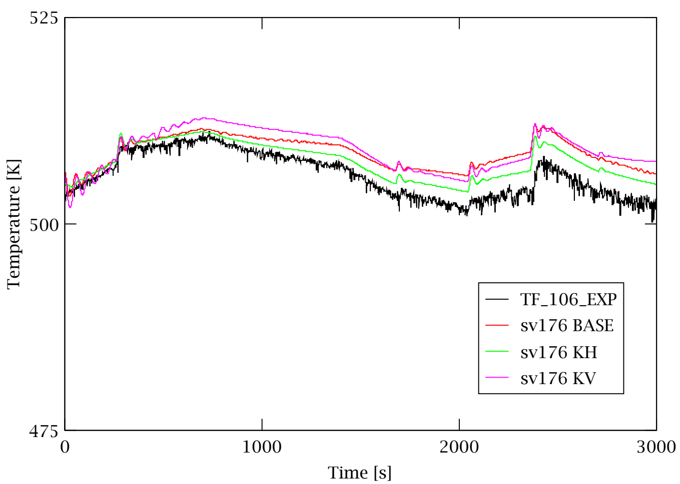
CORE POWER



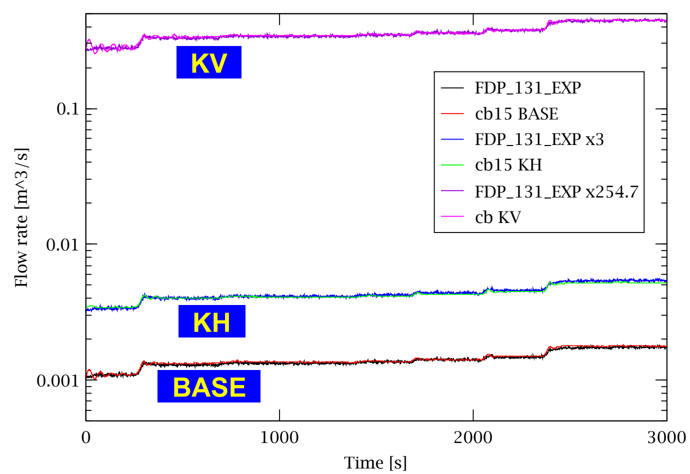
CORE INLET TEMPERATURE



CORE OUTLET TEMPERATURE



VOLUMETRIC PRIMARY FLOW RATE



From a visual observation of the selected parameters, they all result to be qualitatively and quantitatively comparable considering the proper scaling factor.

QUALITATIVE ACCURACY EVALUATION OF THE CALCULATIONS (BASE, KH, KV)

Phenomenon	Experiment		TRACE BASE	TRACE KH	TRACE KV
	Phenomena	Measurement	Phenomena	Phenomena	Phenomena
Single-phase Natural Circulation	+	+	+	+	+
Heat Transfer in Covered Core	+	+	+	+	+
By Pass Heat Transfer	+	+	+	+	+
Distribution of Pressure Drop Through Primary System	+	+	+	+	+
Heat Transfer in SG Primary Side	+	+	+	+	+
Structural Heat and Heat Losses	+	o	+	+	+
Heat Transfer in SG Secondary Side	+	+	+*	+*	+*
Steam Superheated on Secondary Side	+	+	+*	+*	+*

**The curved pipe option has been used for simulating flow through a curved pipe such as secondary side of a helical steam generator.* This option is available from TRACE V5 patch 4. In general the model is able to quantitatively better predict heat transfer between primary and secondary side through an helical coil SG. However still some quantitative difference are present. The quantitative analyses done by FFTBM for the OSU-MASLWR-002 and 003A test show that the quantitative discrepancies are higher when the FW mass flow rate increase.*

	+	o	NA	-
Experimental data	Phenomenon occurred in the test and it is directly measured	Phenomenon occurred in the test and it is indirectly measured	Phenomenon occurred during the test but there is no instrumentation to detect (lack of instrumentation)	Phenomenon not occurred in the test
Calculated data	Phenomenon is clearly predicted by the code **(Excellent/Reasonable)**	Phenomenon is partially predicted (i.e. the answer of the code is **reasonable** but closure code relation are not appropriate, etc)	Models are not appropriate to predict (i.e. nodalization strategy, etc) **(Minimal)**	Phenomenon is not predicted by the code **(Unqualified)**

SCALING ADIMENSIONAL NUMBERS ANALYSES

- In addition to the accuracy evaluation, the code scaling has been assessed considering some relevant characteristic and non-dimensional numbers (IAEA TECDOC-1474, Annex 11):
  - Loop time constant

$$\tau_{loop} = \sum_{i=1}^{N} \frac{l_i}{u_i} = \sum_{i=1}^{N} \tau_i = \frac{M_{sys}}{\dot{m}_o} = \frac{M_{sys}}{\rho_l u_{co} a_c}$$

  - Richardson number

$$\Pi_{Ri} = \frac{\beta g (T_H - T_C)_o L_{th}}{u_{co}^2}$$

  - SG heat transport number

$$\Pi_{SG} = \frac{\dot{q}_{SGo}}{\rho_l u_{co} a_c C_{pl} (T_M - T_C)_o}$$

  - Loop heat loss number

$$\Pi_{Loss} = \frac{\dot{q}_{loss,o}}{\rho_l u_{co} a_c C_{pl} (T_M - T_C)_o}$$

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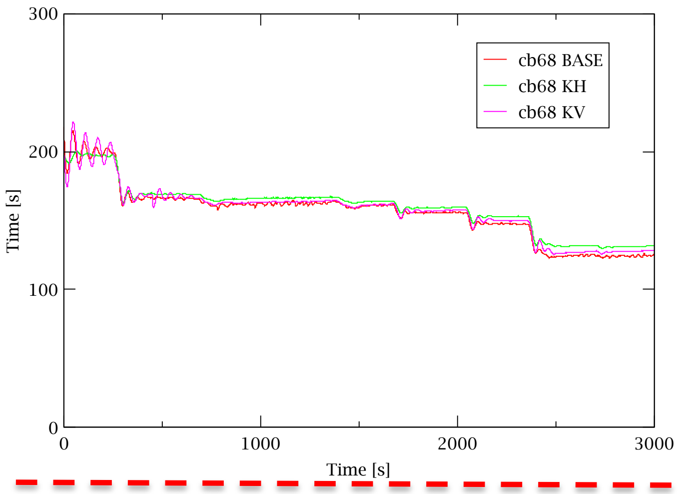
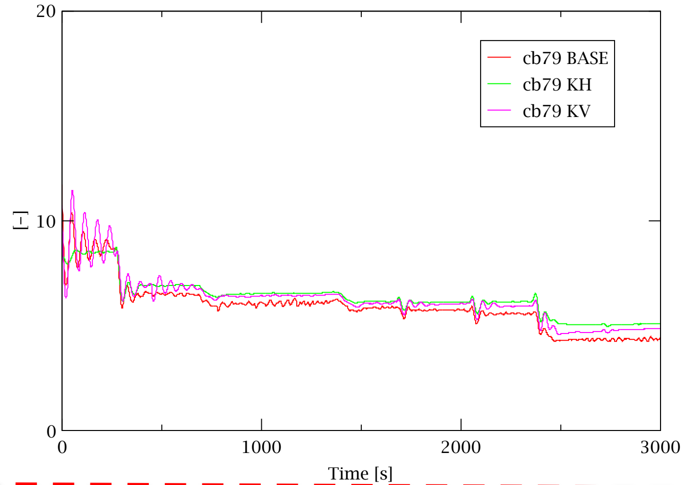
SCALING ADIMENSIONAL NUMBERS ANALYSES

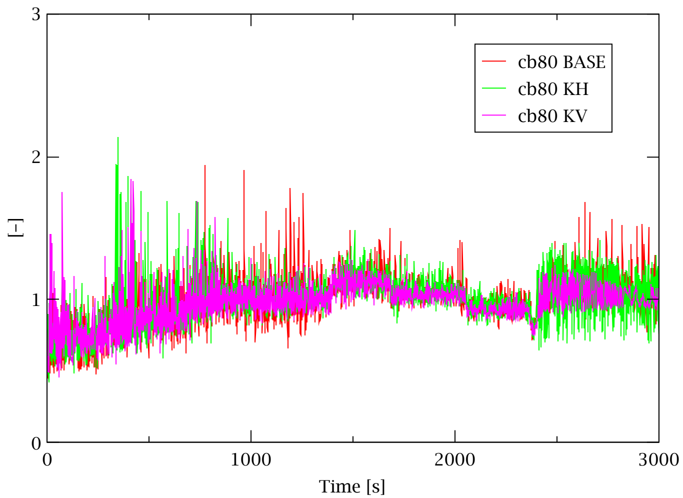
LOOP TIME CONSTANT



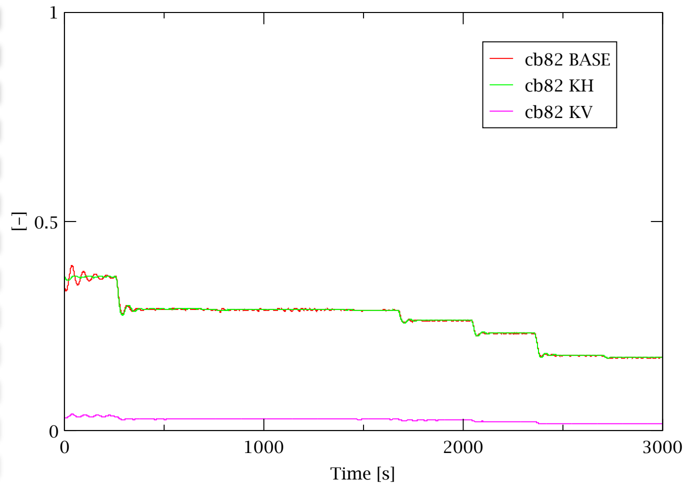
RICHARDSON NUMBER



SG HEAT TRANSPORT NUMBER



LOOP HEAT LOSS NUMBER



From a visual comparison the only number not preserved is the loop heat loss number in the KV calculation. In fact, the outer heat transfer surface (and consequently heat losses) does not scale linearly with the core power.

SCALING ADIMENSIONAL NUMBERS ANALYSES

- The FFTBM has been applied to characterize the discrepancy of the characteristic and non-dimensional numbers between the three models considering the base one as reference.

Variables	Window 0 - 250		Window 250 - 667		Window 667 - 1370		Window 1370 - 2378		Window 2378 - 3060	
	AA_KH	AA_KV	AA_KH	AA_KV	AA_KH	AA_KV	AA_KH	AA_KV	AA_KH	AA_KV
Loop_TC	0.16	0.12	0.06	0.07	0.07	0.06	0.08	0.05	0.10	0.08
Richardson	0.40	0.27	0.15	0.16	0.17	0.16	0.22	0.16	0.23	0.19
SG_number	0.87	1.07	1.16	0.99	0.90	0.85	0.72	0.63	1.00	0.71
Heat_loss_number	0.13	0.90	0.03	0.91	0.05	0.91	0.02	0.90	0.05	0.91
total	0.39	0.59	0.35	0.53	0.30	0.50	0.26	0.44	0.35	0.47

- The loop time constant and the Richardson number have an AA below 0.4.

- The SG heat transport number high AA is due to the oscillations observed in the calculated results with all models*. However, the visual analyses shows that the SG number is consistent in the three calculations.

- The heat loss number relative high AA in the KV calculation is due to the missing preservation of this number as already underlined.

*In relation to the adoption of the FFTBM method it is to underline that the presence of oscillations in the different calculated data could give relatively high values of AA even if the curves, from a visual observation, seem in reasonable agreement. These oscillations in fact can introduce higher frequencies that in principle could be not physical but add spurious contribution in the AA computation increasing its value. Therefore in a validation process for safety review purpose, more detailed analysis could be necessary to analyze the nature of the oscillations both in the experimental and/or calculated signals and by investigating the AA values as a function of the cut-off frequency.

MODELING LESSONS LEARNED

- The scaling analysis of the natural circulation OSU-MASLWR-002 shows that the main phenomena are predicted by the code in the different scaled configurations both qualitatively and quantitatively. Relevant characteristics and non-dimensional numbers are in general preserved between the base and scale-up models.
- Future activities are in progress to extend the scaling-up analysis of the TRACE model and code to the blowdown tests. Also, considering the current development of the code, the new features available to the user will be tested.

SASPAM-SA

SAFETY ANALYSIS OF SMR WITH PASSIVE MITIGATION STRATEGIES - SEVERE ACCIDENT

SASPAM-SA (Safety Analyses of SMR with Passive Mitigation strategies - Severe Accident) Horizon Euratom Project

Fulvio Mascari

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HORIZON EURATOM SASPAM-SA PROJECT

Key Object and Outcomes

SAFETY ANALYSIS OF SMR WITH PASSIVE MITIGATION STRATEGIES - SEVERE ACCIDENT

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

HORIZON EURATOM SASPAM-SA PROJECT

SASPAM-SA

Key Highlights and Consortium

SAFETY ANALYSIS OF SMR WITH PASSIVE MITIGATION STRATEGIES - SEVERE ACCIDENT

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

HORIZON EURATOM SASPAM-SA PROJECT

HORIZON EURATOM SASPAM-SA PROJECT

SASPAM-SA: OPEN WORKSHOPS

SAFETY ANALYSIS OF SMR WITH PASSIVE MITIGATION STRATEGIES - SEVERE ACCIDENT

SASPAM-SA
FIRST OPEN WORKSHOP

INTERNATIONAL WORKSHOP ON SMR SAFETY FOR A SUSTAINABLE SHORT-TERM DEPLOYMENT

OCTOBER 17TH – 18TH, 2024 IRSN HEADQUARTER

31 AV. DE LA DIVISION LECLERC, FONTENAY – AUX – ROSES, FRANCE

The SASPAM-SA open workshop will promote international exchange of information and practices related to light water SMR safety considering the outcomes from the current ongoing Horizon Euratom SASPAM-SA project, other EU-Projects and other international initiatives (e.g. European SMR Pre-partnership, IAEA, OECD/NEA, ETSON, etc.). The workshop's objective is to share the advancements of the current major research activities on light water SMR safety and give the opportunity to discuss the main results. This allows to identify the needed knowledge development, in the view of additional short-term research actions, to support the SMR European licensing process.

>>> FIRST CALL FOR EXTENDED ABSTRACT <<<

PRE-APPLICATION FORM

CONTACT PERSONS

For technical information, please contact:

- Mr. Ahmed Bentaib, IRSN [Tel: +33 1 58 35 98 54, e-mail:ahmed.bentaib@irsn.fr]
- Mr. Fulvio Mascari, ENEA [Tel: +39 3881135591, e-mail: fulvio.mascari@enea.it]

Submission:

- Instructions for Extended Abstract preparation appear in the last page and the template is available at the workshop website: SASPAM-SA / International Open Workshop – SASPAM-SA (on-going)
- Notification of abstract acceptance by email;
- Extended Abstract will be reviewed by the Scientific Committee;
- Oral and posters sessions can be envisaged;
- Selected papers will be encouraged for publication in technical journals.

ATTENTION PLEASE:

Important dates:

Extended Abstract submission: May 31th, 2024

Notification of Abstract acceptance: June 30th, 2024

Final Extended Abstract submission:September 30th, 2024

HORIZON EURATOM SASPAM-SA PROJECT

SASPAM-SA

SASPAM-SA: OPEN WORKSHOPS - ENEN2 plus

group application accepted for young researcher

SAFETY ANALYSIS OF SMR WITH PASSIVE MITIGATION STRATEGIES - SEVERE ACCIDENT

ENEN2Plus Mobility Portal

Accepted: The application 0000000190 has been APPROVED by the ENEN# Mobility Committee.

- Granted number of attendees = 20 (requested = 25)
- Granted amount per attendee = 800 (requested = 800) euros
- Total granted amount = 16000 EUR

M.Sc. students in nuclear interested in extracurricular experience and/or academic exchange such as the EMSNE certification by ENEN

Ph.D. students in nuclear interested in academic and research exchange, access to research infrastructures and cooperation with Euratom research projects

Post-docs in nuclear interested in academic and research exchange, access to research infrastructures and cooperation with Euratom research projects

Early career professionals (up to 10 years of experience) interested in changing their careers to nuclear and/or building up their life-long learning on nuclear topics

HORIZON EURATOM SASPAM-SA PROJECT

SASPAM-SA

Acknowledgement/Disclaimer

SAFETY ANALYSIS OF SMR WITH PASSIVE
MITIGATION STRATEGIES - SEVERE ACCIDENT

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Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or European Commission-Euratom.
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FOR PSI Project funded by

Schweizerische Eidgenossenschaft
Confédération suisse
Confederazione Svizzera
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