Overview and Main Results of the IAEA CRP I31033 “Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water Cooled Reactors”

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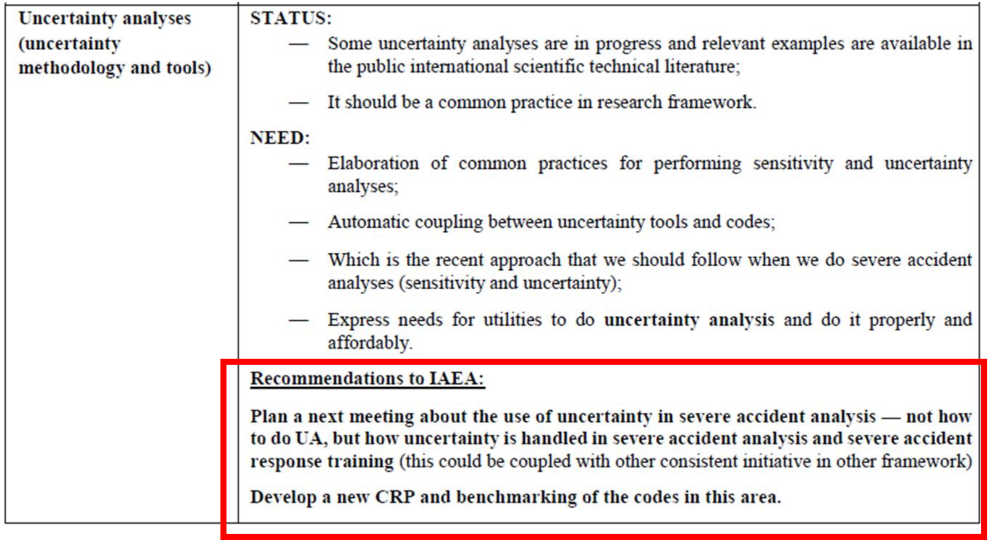
Institute for Neutron Physics and Reactor Technology

Backgroung

- **CRP rationale** ← Observations from 2017 IAEA TM (IAEA TECDOC 1872)
- *Severe accident (SA) 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.*



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.**

IAEA TECDOC 1872, https://www-pub.iaea.org/MTCD/Publications/PDF/TE-1872web.pdf

The CRP I31033 (2019-2024): Objectives

Karlsruhe Institute of Technology

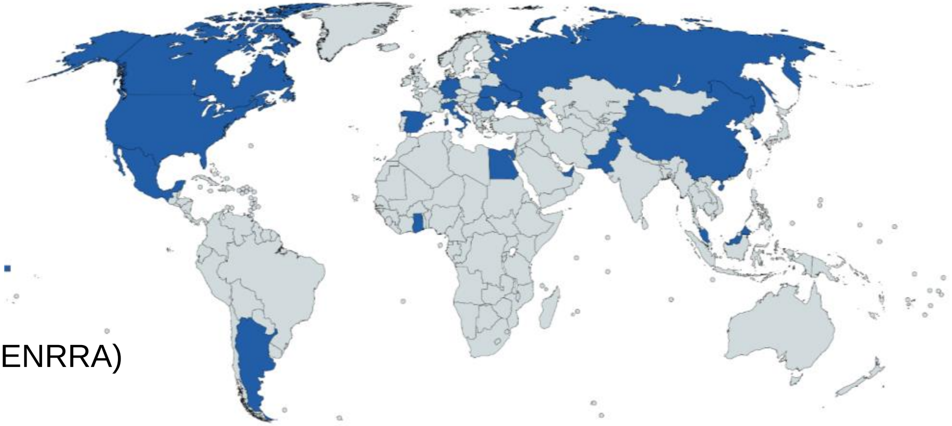
- **Bring together the current state-of-knowledge on uncertainty propagation in SA analyses accumulated by experienced analysts**
- Achieve significant improvement in sophistication and quality of SA analyses performed by the participants from Member States with well developed knowledge, adequate simulation capabilities, and long years of relevant practice
- Enable objective peer review of the benchmark studies with various SA codes by the participating Member States → **new knowledge and sharing of research results relevant to evaluation of U&SA of SA simulations and modelling**
- **Foster national excellence and international cooperation** through an exercise to elevate the capability and sophistication of global severe accident code users
- **Promote sharing of newly developed knowledge and contribute to capacity building in developing countries**

The CRP I31033: Outcomes

- **Improving the capabilities and the expertise in Member States to perform SA analysis and related U&SA for Water Cooled Reactors**

1. **Development, testing, and application of U&S methodologies and of Severe Accident/Uncertainty Tools (SA/UT) ↔ Foster a common understanding of U&S methodologies and tools among Member States**
2. **Establish best practise for U&SA in the realm of SA analysis**
3. **Large number of uncertainty parameters (UP) identified and characterized for each technology addressed in the CRP (PWR, iPWR, VVER, CANDU, BWR Mark-I and -II)**
4. **Large database assessed relevant accident sequences**
5. **Dissemination and education initiatives**

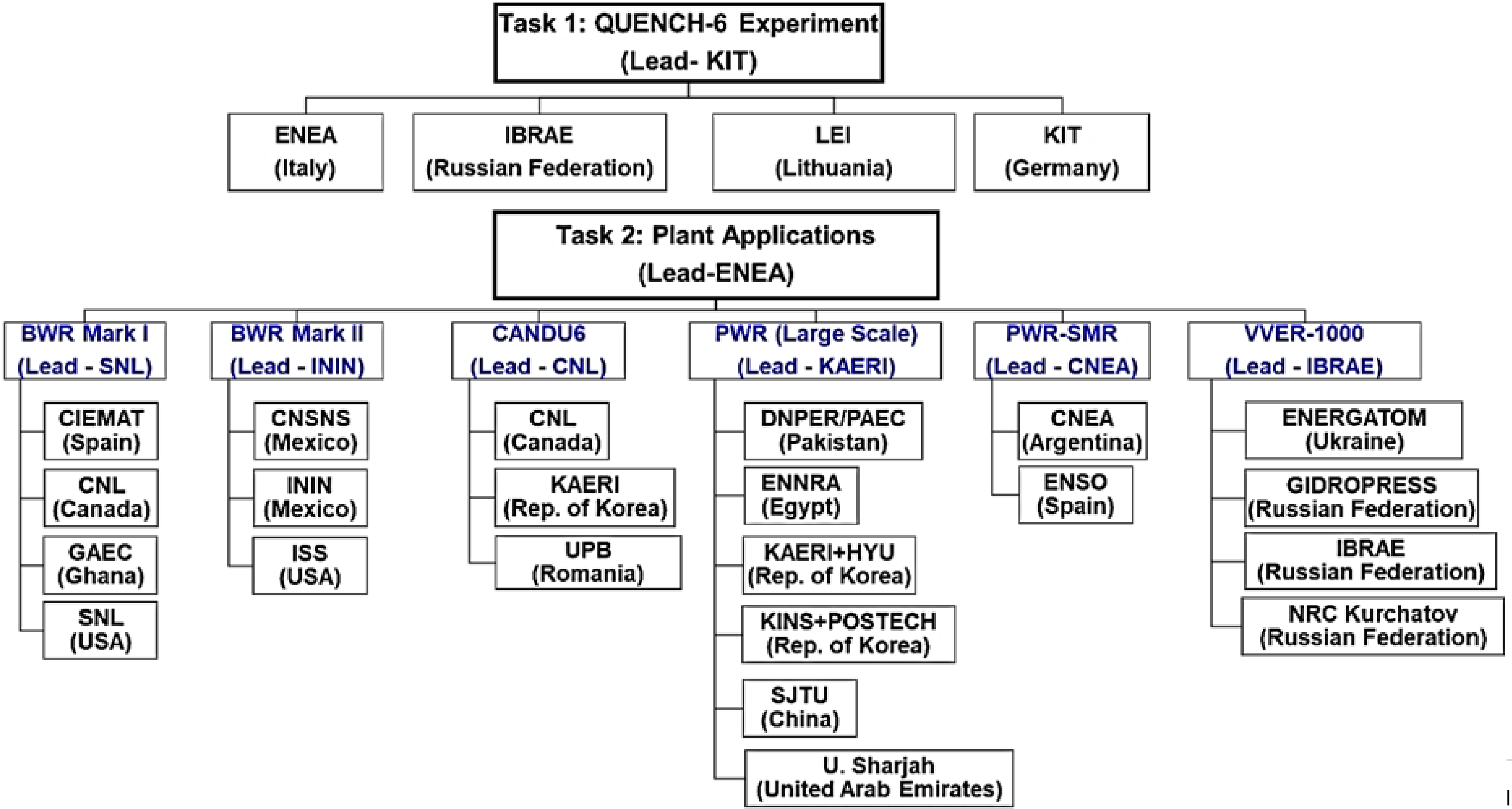
CRP Participants



- 22 organization from 18 Member States

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)
Ghana: Ghana Atomic Energy Commission (GAEC)
Italy: Agency for New Technology, Energy and Sustainable Economic Development (ENEA)
Lithuania: Lithuanian Energy Institute (LEI)
Malaysia: Malaysian Nuclear Agency (MNA)
Mexico: National Institute for Nuclear Studies (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)
Ukraine: Scientific and technical Center of SE NNEGC Energoatom
United Arab Emirates: University of Sharjah
United States of America: Sandia National Laboratory (SNL) , Innovative Systems Software (ISS)

The CRP I31033: Tasks and Sharing



- Task 1: QUENCH-6 Experiment (Lead- KIT)
  - ENEA (Italy)
  - IBRAE (Russian Federation)
  - LEI (Lithuania)
  - KIT (Germany)
- Task 2: Plant Applications (Lead-ENEA)
  - BWR Mark I (Lead - SNL)
    - CIEMAT (Spain)
    - CNL (Canada)
    - GAEC (Ghana)
    - SNL (USA)
  - BWR Mark II (Lead - ININ)
    - CNSNS (Mexico)
    - ININ (Mexico)
    - ISS (USA)
  - CANDU6 (Lead - CNL)
    - CNL (Canada)
    - KAERI (Rep. of Korea)
    - UPB (Romania)
  - PWR (Large Scale) (Lead - KAERI)
    - DNPER/PAEC (Pakistan)
    - ENNRA (Egypt)
    - KAERI+HYU (Rep. of Korea)
    - KINS+POSTECH (Rep. of Korea)
    - SJTU (China)
    - U. Sharjah (United Arab Emirates)
  - PWR-SMR (Lead - CNEA)
    - CNEA (Argentina)
    - ENSO (Spain)
  - VVER-1000 (Lead - IBRAE)
    - ENERGATOM (Ukraine)
    - GIDROPRESS (Russian Federation)
    - IBRAE (Russian Federation)
    - NRC Kurchatov (Russian Federation)

Task 1

QUENCH-06 test application uncertainty exercise

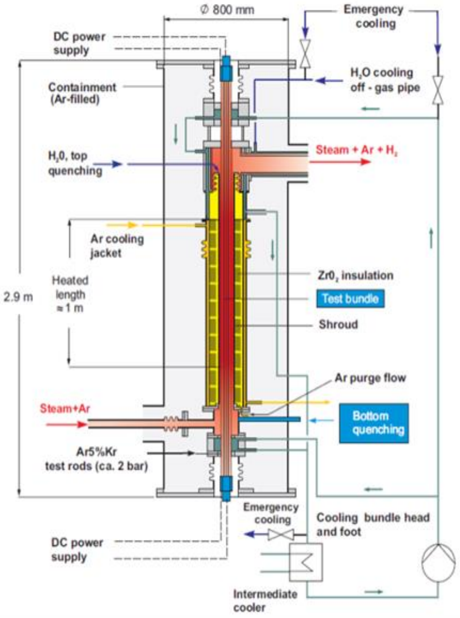
Motivation

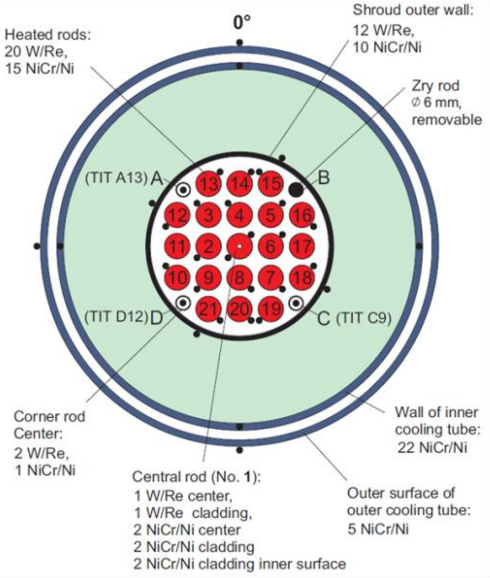
- **Testing relevant calculation platforms of SA/UT in view of their applications to plant analyses supporting other CRP tasks**
- The QUENCH-06 test (KIT, December 2000) selected as test case:
  - large amount of available qualified experimental data
  - special importance for code validation → OECD International Standard Problem (ISP) no. 45

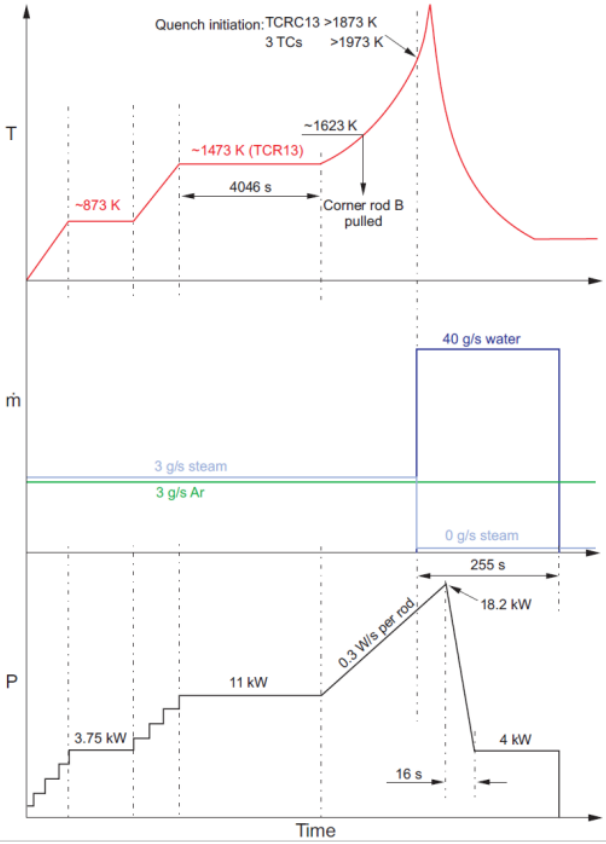
Institution	**Severe Accident Code**	**Uncertainty Code**
ENEA (Italy)	ASTEC	RAVEN
IBRAE (Russian Federation)	SOCRAT	ELENA
KIT (Germany)	ASTEC	URANIE and KATUSA
LEI (Lithuania)	RELAP/SCDAPSIM	SUSA

The QUENCH-06 Test

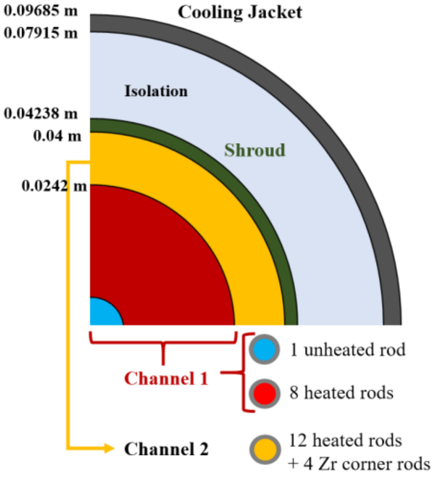
- **Goal**: investigating the coolability and determining the hydrogen source term in a PWR pre-oxidized rod bundle quenched with water → insight on SAMGs

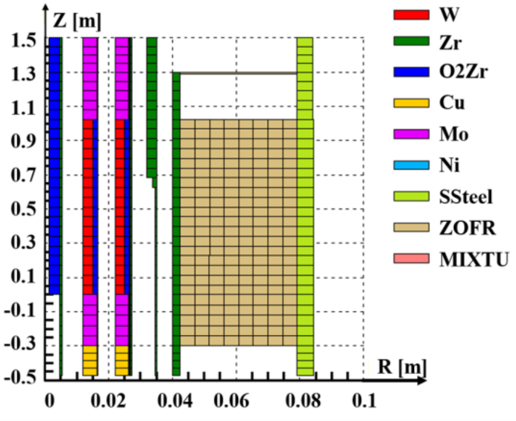


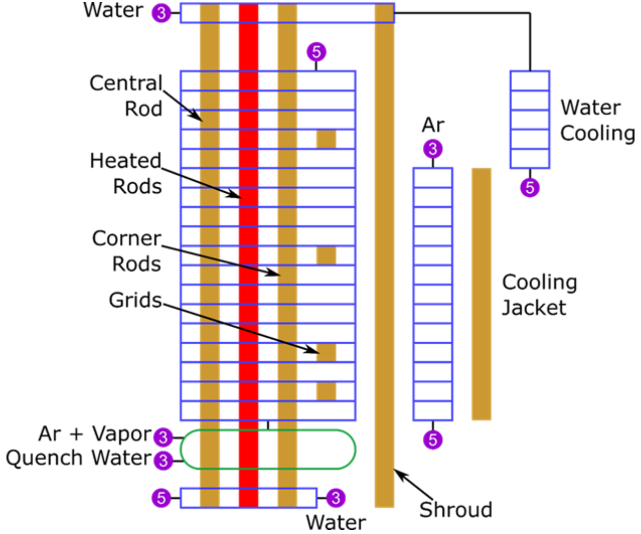


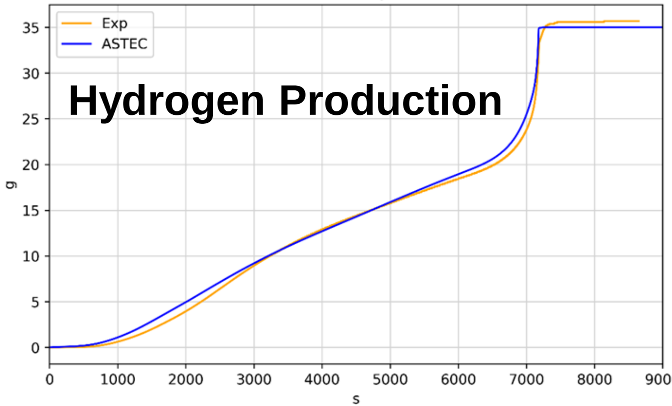


Modelling and Validation

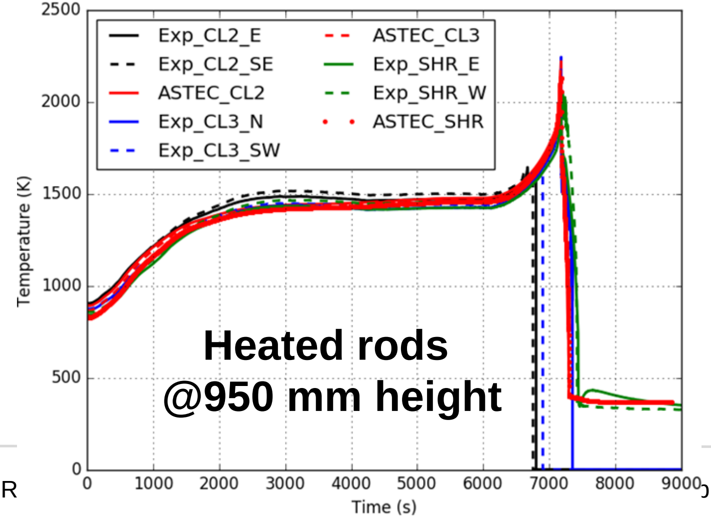








Hydrogen Production



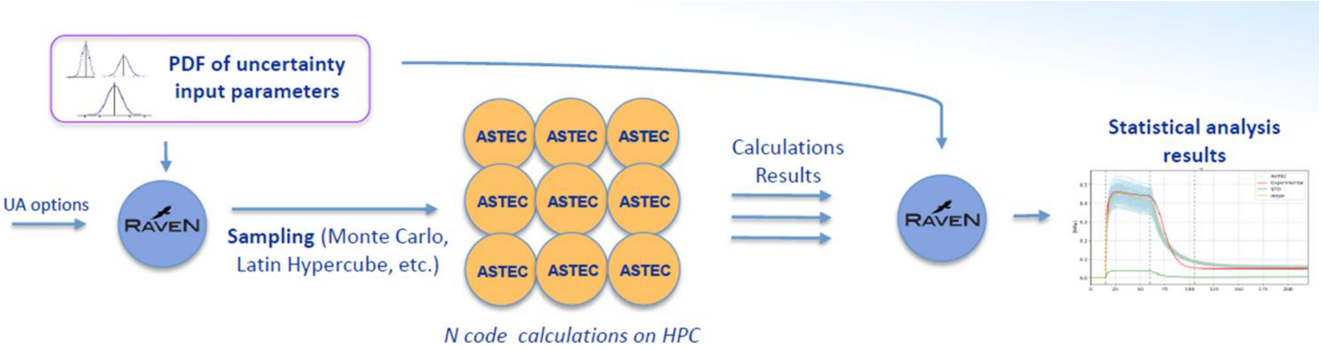
Heated rods
@950 mm height

The U&SA Exercise

- **23 uncertainty input parameters related to**:
  - Geometry of the Bundle (6)
  - Boundary Conditions (8)
  - Integrity criteria (2)
  - Radiative Heat Transfer modelling (2)
  - Convection Heat Transfer modelling (3)
  - Material Movement Model (2)
- **FoMs**
  - Total mass of accumulated Hydrogen
  - Temperature of the central fuel rod simulator at elevation 950 mm
  - Hydrogen generation rate
  - Axial profile of the oxide scale of the corner rod at 6620 s
  - Axial profile of the average oxide scale of the heated rods at the end of the scenario

SA code/U&SA tool Coupling and Methodologies (1/2)

ASTEC/RAVEN



PDF of uncertainty input parameters

UA options

RAVEN

Sampling (Monte Carlo, Latin Hypercube, etc.)

ASTEC ASTEC ASTEC
ASTEC ASTEC ASTEC
ASTEC ASTEC ASTEC

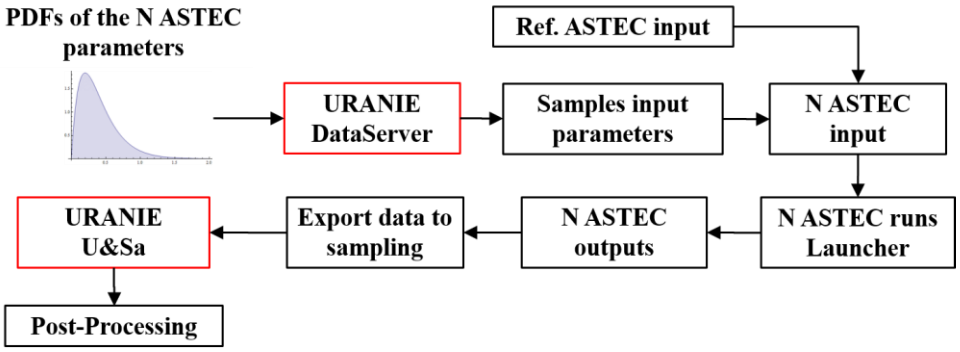
N code calculations on HPC

Calculations Results

RAVEN

Statistical analysis results

ASTEC/URANIE



PDFs of the N ASTEC parameters

URANIE DataServer

Samples input parameters

Ref. ASTEC input

N ASTEC input

N ASTEC runs Launcher

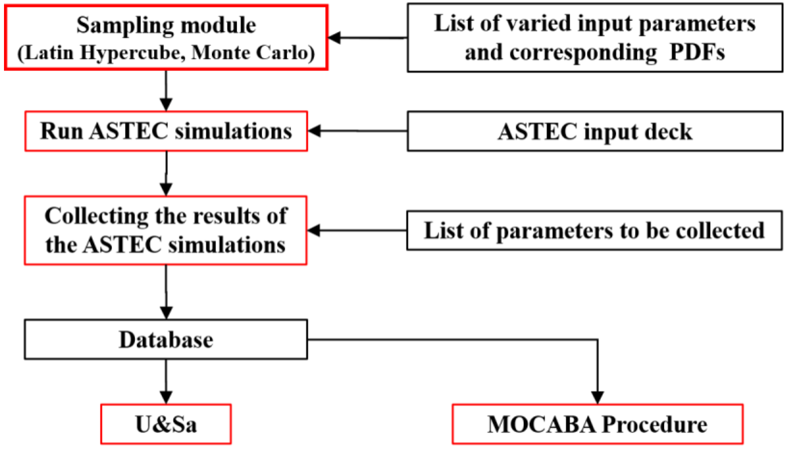
N ASTEC outputs

Export data to sampling

URANIE U&Sa

Post-Processing

ASTEC/KATUSA



Sampling module (Latin Hypercube, Monte Carlo)

List of varied input parameters and corresponding PDFs

Run ASTEC simulations

ASTEC input deck

Collecting the results of the ASTEC simulations

List of parameters to be collected

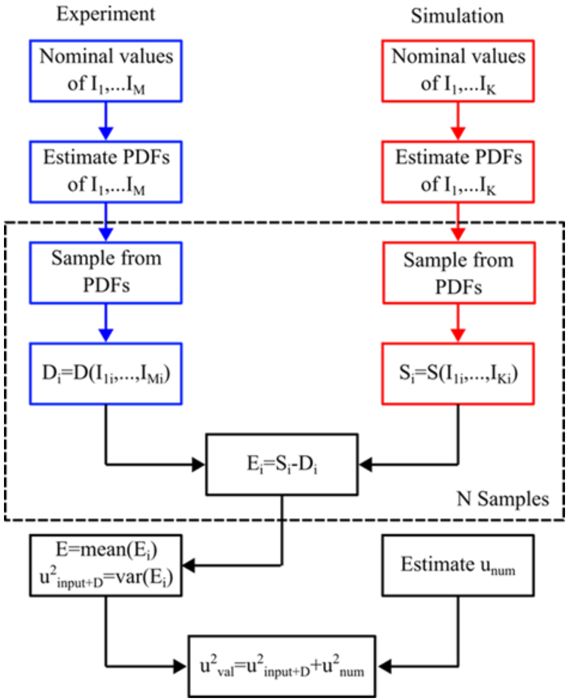
Database

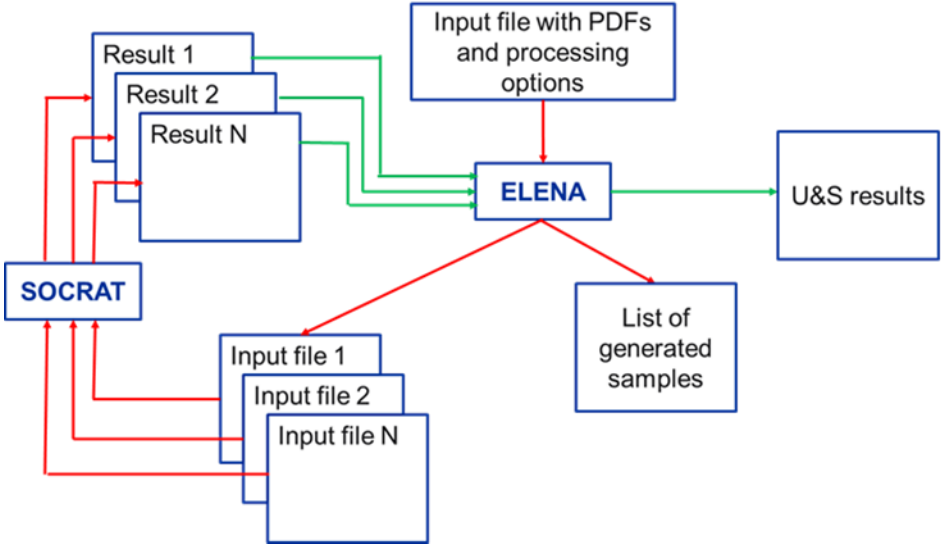
U&Sa

MOCABA Procedure

SA code/U&SA tool Coupling and Methodologies (2/2)

- Development and application of a MC method taking into account the model error, the numerical error, and the input data error (IBRAE RAN)





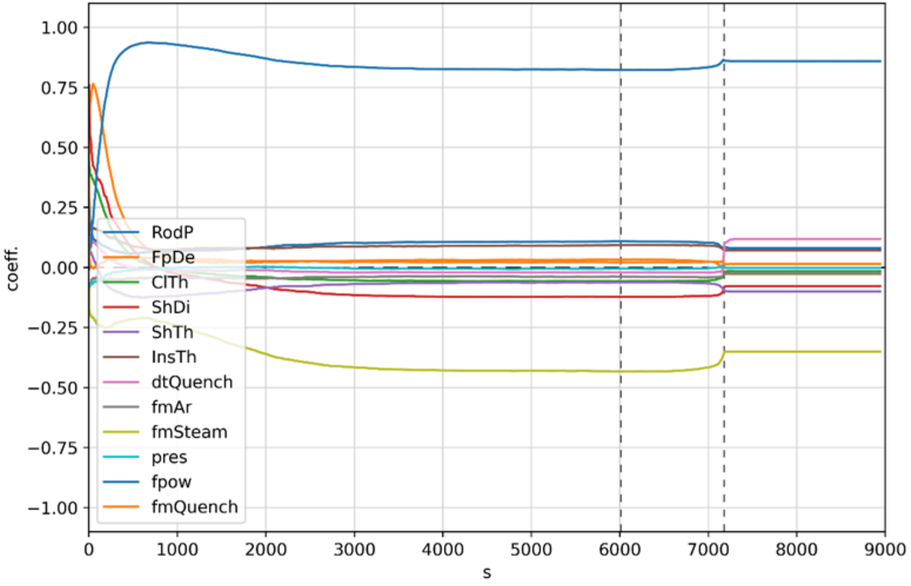
Methodology for U&SA

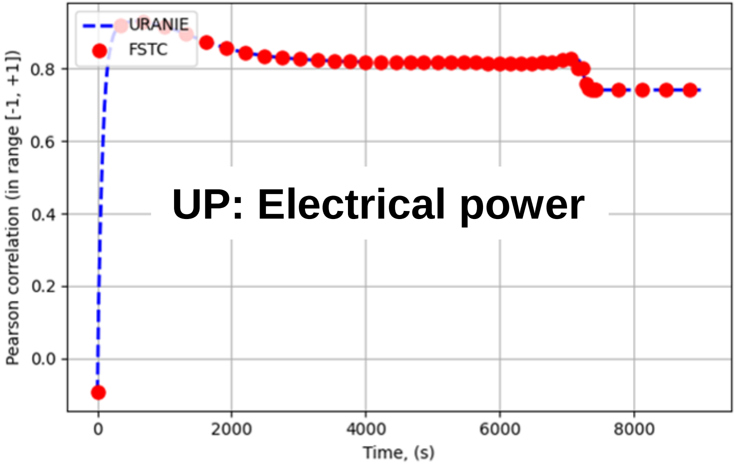
Institution	Sampling and UQ method	Sensitivity/importance analysis
ENEA (Italy)	SRS **N = 200**	Pearson and Spearman
IBRAE (Russian Federation)	SRS **N = 1000**	Sobol
KIT (Germany)	LHS **N = 600**	Pearson and Spearman
LEI (Lithuania)	SRS GRS Methodology	Pearson and Spearman

U&SA: Results

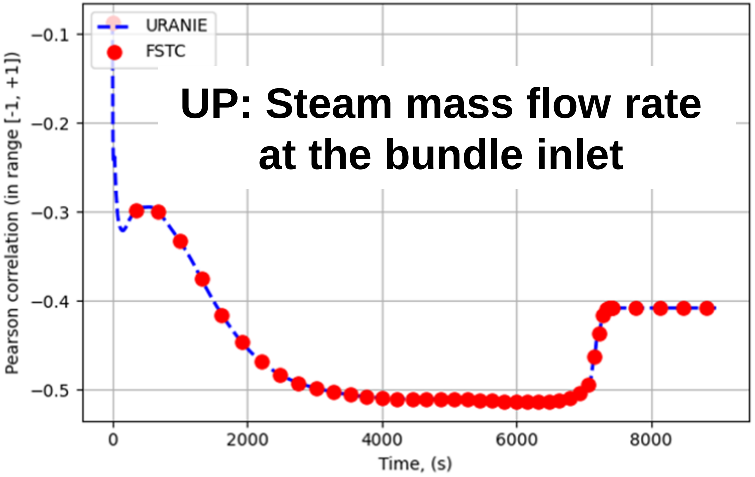
- ➢ FOM: total mass of accumulated Hydrogen

Spearman Correlation (ASTEC/RAVEN)





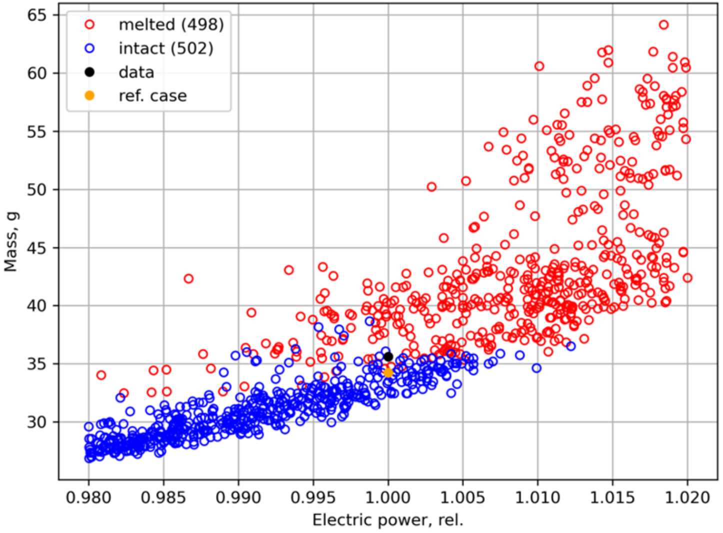
UP: Electrical power



UP: Steam mass flow rate at the bundle inlet

U&SA: Results

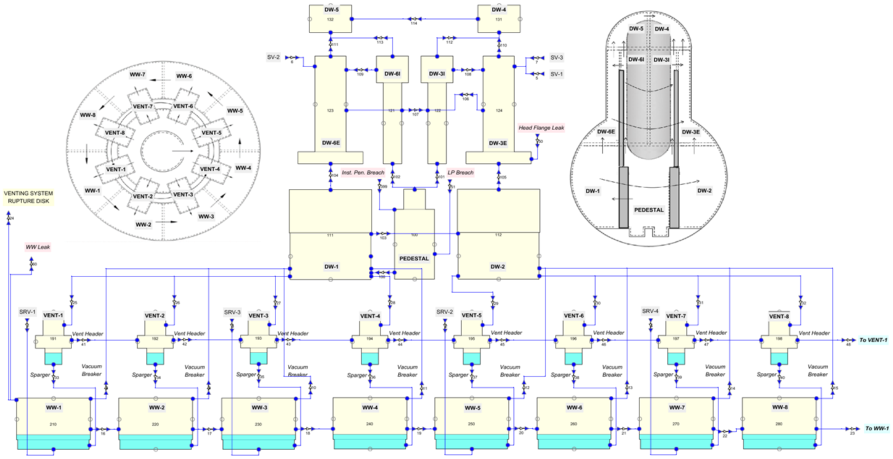
- SOCRAT/ELENA: 1000 calculation run

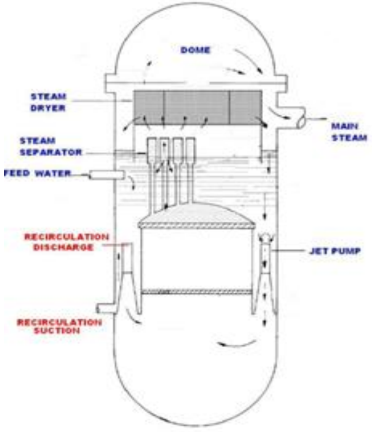


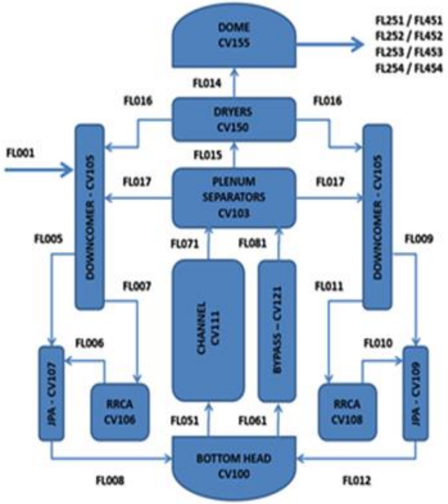
FOM	Parameter
1	Total mass of accumulated hydrogen
2	Mass of Hydrogen accumulated by quenching
3	Mass of Hydrogen accumulated at quenching
4	Peak Hydrogen production rate

FOM #	$\overline{S}$	$\overline{D}$	$\overline{E}$, %	$u_{input+D}$, %	u_{num}, %	u_{val}, %
1	37.1 g	35.7 g	4.0	22.3	6.8	23.3
2	34.5 g	31.9 g	8.2	17.6	3.9	18.0
3	2.6 g	3.8 g	-31.2	71.7	52.2	88.7
4	247 mg/s	236 mg/s	4.5	52.8	6.8	53.2

Task 2 - BWRs


DW-5
DW-4
DW-6I
DW-3I
DW-6E
DW-3E
SV-2
SV-3
SV-1
Head Flange Leak
Inst. Pen. Breach
LP Breach
DW-1
DW-2
PEDESTAL
VENTING SYSTEM RUPTURE DISK
WW Leak
WW-1
WW-2
WW-3
WW-4
WW-5
WW-6
WW-7
WW-8
VENT-1
VENT-2
VENT-3
VENT-4
VENT-5
VENT-6
VENT-7
VENT-8
SRV-1
SRV-2
SRV-3
SRV-4
Vent Header
Vacuum Breaker
Sparger
To VENT-1
To WW-1


DOME
STEAM DRYER
STEAM SEPARATOR
FEED WATER
MAIN STEAM
RECIRCULATION DISCHARGE
JET PUMP
RECIRCULATION SUCTION


DOME CV155
FL251 / FL451
FL252 / FL452
FL253 / FL453
FL254 / FL454
FL014
FL016
FL016
DRYERS CV150
FL001
FL015
DOWNCOMER - CV105
DOWNCOMER - CV105
FL017
FL017
PLENUM SEPARATORS CV103
FL005
FL071
FL081
FL009
FL007
FL011
CHANNEL CV111
BYPASS – CV121
FL006
FL010
JPA - CV107
RRCA CV106
RRCA CV108
JPA - CV109
FL051
FL061
BOTTOM HEAD CV100
FL008
FL012

Scenarios, Scope, and SA Tools

Plant Type	Institution	Scenario	SA code	Framework of analysis
BWR/3 with Mark I primary containment	GAEC (Ghana)	**Unmitigated SBO** **(In-vessel)**	MELCOR2.2	SAM support
	CIEMAT (Spain)	**SBO** **(In-vessel)**	MELCOR	External uncertainties affecting Source Term
	SNL (USA)	**SBO** **(In-/Ex-vessel)**	MELCOR V2.2r15348	Model development
BWR/5 with a Mark II	CNSNS (Mexico)	**SBO with RCIC injection and ADS actuation** **(In-vessel)**	MELCOR2.1	Support technical basis on regulation issues
	ININ (Mexico)	**Unmitigated high pressure SBO** **(In-vessel)**	MAAP 5.03	Support development and review of technical basis of SAMGs

Methodology for U&SA

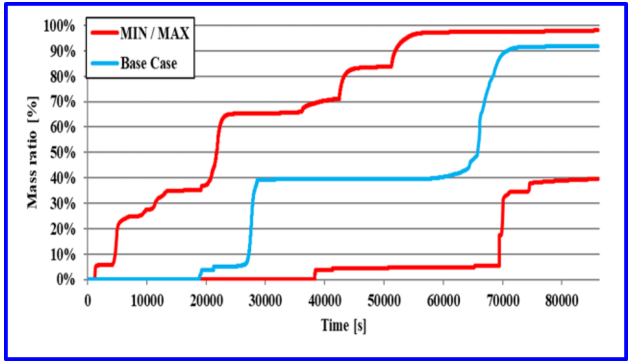
Plant Type	Institution	Sampling and UQ method	UQ tool and calculation scheme	Sensitivity/importance analysis
BWR/3 with Mark I primary containment	GAEC (Ghana)	LHS (two-sided Wilks, 95%/95%)	DAKOTA	Pearson and Spearman
	CIEMAT (Spain)	SRS (two-sided Wilks, 95%/95%)	DAKOTA	Pearson and Spearman
	SNL (USA)	-	In-house	SNL Methodology
BWR/5 with a Mark II	CNSNS (Mexico)	SRS (two-sided Wilks, 95%/95%)	DAKOTA	Pearson and Spearman
	ININ (Mexico)	SRS (two-sided Wilks, 95%/95%) + Sensitivity Analysis Monte-Carlo Filtering Technique	AZTUSIA + other in-house tools	Pearson, Spearman, Partial, Partial Rank

UP and FOMs

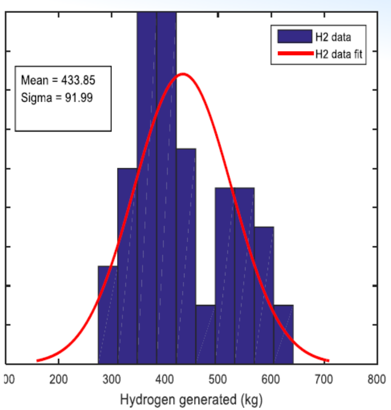
Plant Type	Institution	UP	FOM
BWR/3 with Mark I primary containment	GAEC (Ghana)	**11** (MELCOR2.2)	In-vessel **H2 generation**
	CIEMAT (Spain)	**150** (MELCOR)	1. Noble gases, I, Cs 2. **Total amount of FPs released** 3. Onset of fission product release 4. Fission product release rates 5. End time of fission product release
	SNL (USA)	**14** (MELCOR2.2)	1. **Overall accident Progression** 2. H2 generation 3. **Thermal hydraulic response** 4. **Reactor core degradation** 5. RPV lower head breach
BWR/5 with a Mark II	CNSNS (Mexico)	**6** (MELCOR2.1)	**Containment failure time**
	ININ (Mexico)	**28** (MAAP 5.03)	1. **H2 mass** 2. Fission product mass fractions 3. **Core damage criterion time** 4. **Core support plate failure time** 5. Debris mass in lower head 6. RPV breach time

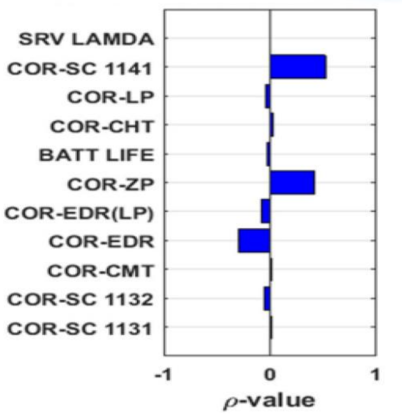
BWR/3 Mark-I: Results

Cs Release from Fuel



- ➢ CIEMAT
  - ➢ The distribution in MELCOR RN model of key radionuclides.
  - ➢ The parametric setting of key phenomena (cladding mechanical temperature).
  - ➢ The Th modelling during the early phase of the accident

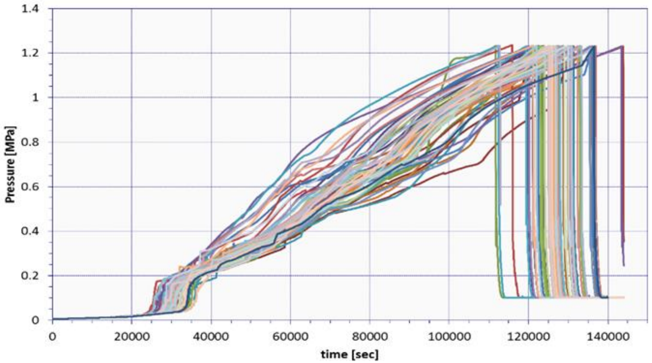


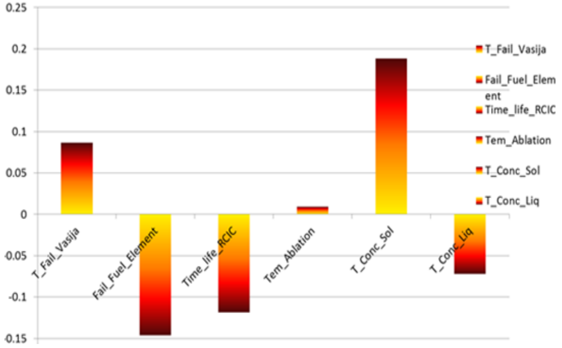


- ➢ GAEC
  - ➢ uncertainty in hydrogen generated ~22.46%
  - ➢ COR_SC 1141 (Core Melt Breakthrough Candling Parameters: Maximum melt flow rate per unit width after breakthrough)
  - ➢ COR_ZP (Porosity of fuel debris beds)
  - ➢ COR_EDR (Particulate debris equivalent diameter in the lower plenum)

BWR/5 Mark-II: Results

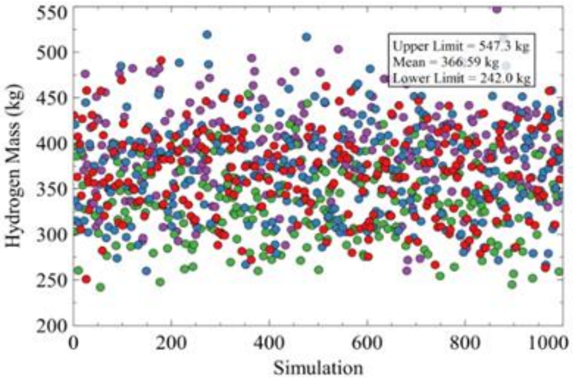
- CSNS

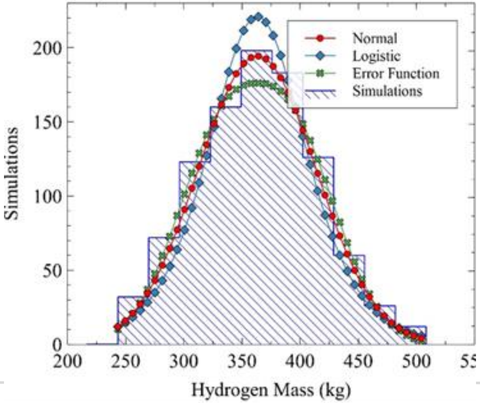




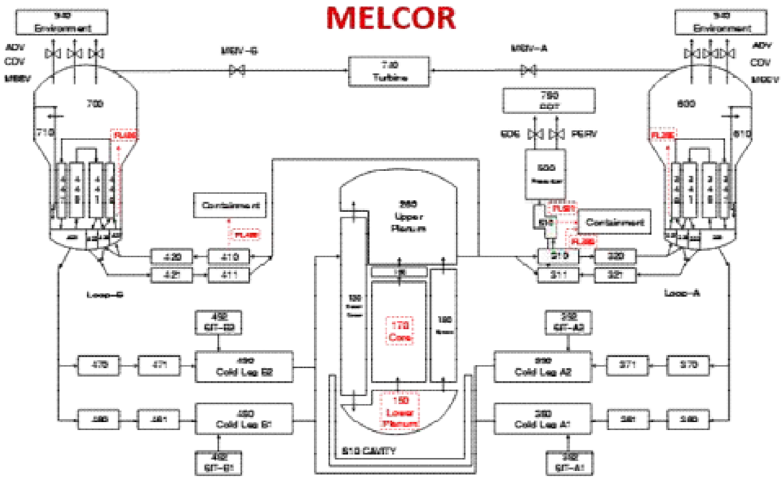
Primary Containment Failure Time Interval [hours]	
Conservative [μ-2σ, μ+2σ]	[31.7, 39]
Best [μ-2.5σ, μ+2.5σ]	[31, 40]

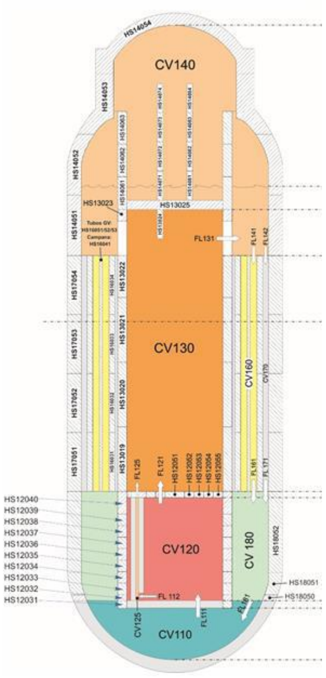
- ININ

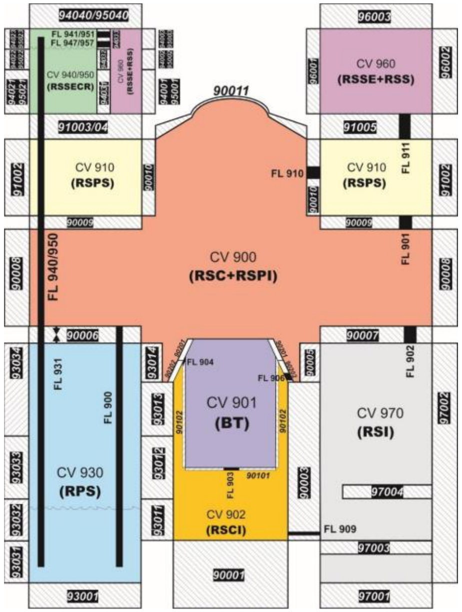




Task 2 – PWR and iPWRs







Scenarios, Scope, and SA Tools

Plant Type	Institution	Reference Plant	Scenario	SA code	Framework of analysis
Large-scale PWR	DNPER (Pakistan)	ACP1000 (K-2 NPP)	SBO (In-/Ex-vessel)	MELCOR1.8.6	Regulatory review support
	ENRRA (Egypt)	KWU-PWR1300	LBLOCA w/o SCRAM (In-vessel)	ATHLET & SCALE6.3	Regulatory review support
	KAERI (Rep. Korea)	OPR1000	STSBO (In-/Ex-vessel)	MELCOR2.2 & MAAP5	SAM & Level 2 PSA support
	KINS (Rep. Korea)	APR1400	SBO (In-vessel & Reactor cavity)	MELCOR2.2 & COOLAP2	Regulatory/safety review support
	SJTU (China)	CPR600	SBO (In-/Ex-vessel)	MELCOR1.8.5	SAM support
	UoS (UAE)	APR1400	SBO (In-vessel, early phase fuel temperature response)	RELAP5/NESTLE-based 3Keymaster simulator	Reactor design and simulation
iPWR	CNEA (Argentina)	CAREM-like Integral Type PWR	SBLOCA (In-vessel)	MELCOR1.8.6	SAM support
	ENSO (Spain)		SBO (In-vessel)	RELAP/SCDAPSIM/MOD3.5	SAM support

Methodology for U&SA

Plant Type	Institution	Sampling and UQ method	UQ tool and calculation scheme	Sensitivity/importance analysis
Large-scale PWR	DNPER (Pakistan)	SRS **N = 2548**	DST (MATLAB –based, In-house)	Pearson, Spearman, and Kendall correlation coefficients
	ENRRA (Egypt)	SVD/UT and LRA approach (SRS: **N = 150**, multivariate sampling)	PYTHON (In-house)	Generalized perturbation theory-based deterministic method
	KAERI (Rep. Korea)	SRS **N = 200**	DAKOTA (MELCOR) / MOSAIQUE (MAAP5, In-house)	Pearson and Spearman correlation coefficients, and PRCC/SRRC
	KINS (Rep. Korea)	LHS **N = 300**	DAKOTA + In-house	Parametric sensitivity analysis
	SJTU (China)	LHS **N = 120**	MATLAB (In-house)	Pearson and Spearman correlation coefficients, and PCC/PRCC
	UoS (UAE)	SRS **N = 120** (perturbation of the parameter space)	DAKOTA + ROMUSE (In-house)	PSA/PCA-based sensitivity
iPWR	CNEA (Argentina)	SRS **N = 59** (one-side Wilks tolerance limit, 95%/95%)	DAKOTA + In-house	Pearson and Spearman coefficients
	ENSO (Spain)		IUA Package	Pearson, Spearman, and Kendall correlation coefficients

UP and FOMs (1/2)

Plant Type	Institution	UP	UP's Domain
Large-scale PWR	DNPER (Pakistan)	**26** (MELCOR1.8.6)	In-/ex-vessel and FPs
	ENRRA (Egypt)	**14** (ATHLET + SCALE6.3)	Coolant void reactivity at different coolant densities, in-vessel
	KAERI (Rep. Korea)	**26** (MELCOR2.2) **29** (MAAP5.05)	In-/ex-vessel and FPs
	KINS (Rep. Korea)	**5** (MELCOR2.2) **8** (COOLAP2)	In-vessel Ex-vessel reactor cavity
	SJTU (China)	**18** (MELCOR1.8.5)	In-vessel
	UoS (UAE)	**44** (3KEYMASTER simulator, 44 groups the SCALE covariance library)	In-vessel
iPWR	CNEA (Argentina)	**10 + 1** (accident management) (MELCOR1.8.6)	In-vessel
	ENSO (Spain)	**15 + 3** (safety systems) (RELAP/SCDAPSIM/MOD3.5)	In-vessel

UP and FOMs (2/2)

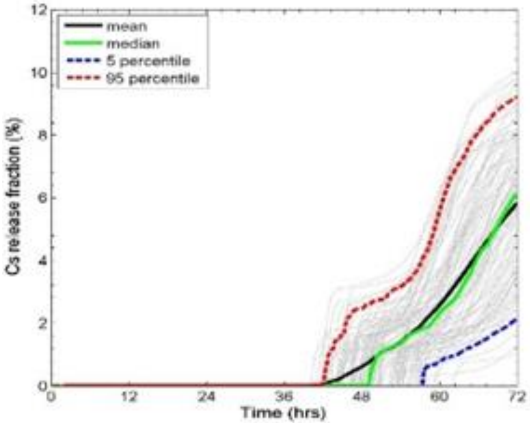
Plant Type	Institution	FOM
Large-scale PWR	DNPER (Pakistan)	1) **Core uncovery time**, 2) RPV failure time, 3) **H_2 in the in-vessel**, 4) $H_2/CO/CO_2$ in the ex-vessel, 5) Containment breach time, 6) **CsI/Cs/Activity release to environment**
	ENRRA (Egypt)	1) **Peak cladding temperature (PCT)**
	KAERI (Rep. Korea)	1) **Core uncovery time**, 2) **RPV lower head (LH) failure time**, 3) Reactor/ containment building (R/B) failure time, 4) H_2/CO in the in-/ex-vessel, 5) Cs release to the environment
	KINS (Rep. Korea)	**1)** **Containment pressure**, 2) **Depth of cavity concrete ablation**, 2) 3) H_2/CO in the in-/ex-vessel
	SJTU (China)	1) **Generation of H_2 in the in-/ex-vessel**
	UoS (UAE)	1) **Early phase fuel temperature (in-vessel)**
iPWR	CNEA (Argentina)	1) **Core uncovery time**, 2) Onset of core degradation, 3) **Core relocation time to the lower plenum**
	ENSO (Spain)	1) **Time to oxidation > 0.1% of the nominal power**, 2) **Time to $T_{cladding}$> 1477 K**, 3) Fuel rupture/ Debris formation/Core slumping/Creep rupture times, 4) **Cumulative H2 generation**, 5) $T_{cladding}$ when fuel rupture, 6) Cumulative fission product (FP), NC, and FP soluble

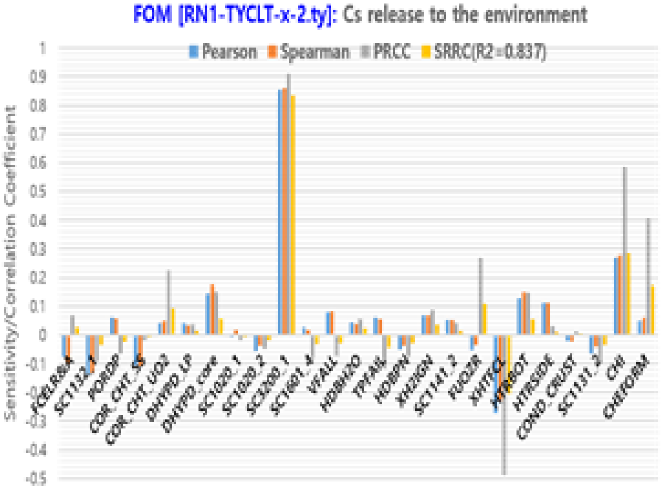
OPR1000 (KAERI): Results

Case ID	RCS Pre-bleed[1]	Feed & Bleed Operation
Base Case	N/A	N/A
Mitigation Case 1	Open of 2 SDS valves at the SAMG entry condition	Mobile pump starts at 4 h to inject external water into RCS
Mitigation Case 2		Mobile pump starts at 4 h to inject external water into SG

[1] RCS pre-bleed via SDS (safety depressurization system) intended to allow utilization of the cooling water inventory of the SITs (safety injection tanks)

- ➢ **Base Case: MELCOR**





CAREM-Like: Results

- **CNEA: LOCA** (failure of all heat removal and injection systems, except for the reactor pressure vessel external cooling system)

	Core uncovery time	Onset of core degradation	Core relocation time to the lower plenum
Max	3.97 h	6.78 h	16.87 h
Min	2.44 h	4.83 h	10.93 h

- **ENSO: SBO**

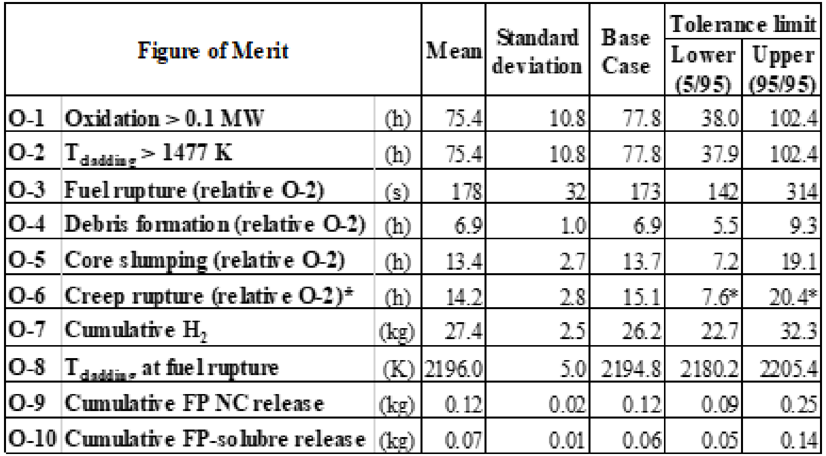
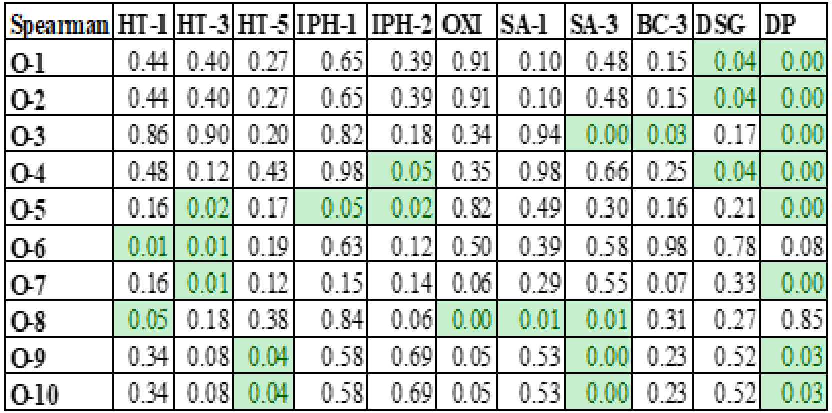


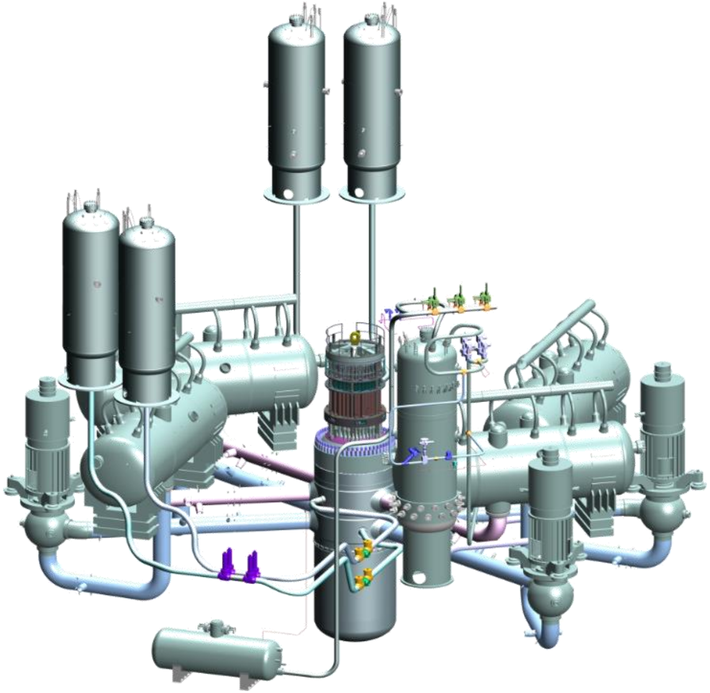
Figure of Merit			Mean	Standard deviation	Base Case	Tolerance limit	
						Lower (5/95)	Upper (95/95)
O-1	Oxidation > 0.1 MW	(h)	75.4	10.8	77.8	38.0	102.4
O-2	$T_{cladding}$ > 1477 K	(h)	75.4	10.8	77.8	37.9	102.4
O-3	Fuel rupture (relative O-2)	(s)	178	32	173	142	314
O-4	Debris formation (relative O-2)	(h)	6.9	1.0	6.9	5.5	9.3
O-5	Core slumping (relative O-2)	(h)	13.4	2.7	13.7	7.2	19.1
O-6	Creep rupture (relative O-2)*	(h)	14.2	2.8	15.1	7.6*	20.4*
O-7	Cumulative H_2	(kg)	27.4	2.5	26.2	22.7	32.3
O-8	$T_{cladding}$ at fuel rupture	(K)	2196.0	5.0	2194.8	2180.2	2205.4
O-9	Cumulative FP NC release	(kg)	0.12	0.02	0.12	0.09	0.25
O-10	Cumulative FP-soluble release	(kg)	0.07	0.01	0.06	0.05	0.14

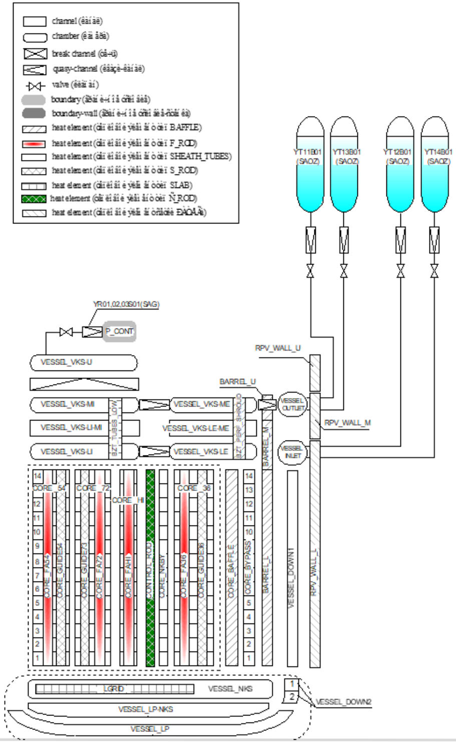
(*) qualitative -only cases with creep rupture



Spearman	HT-1	HT-3	HT-5	IPH-1	IPH-2	OXI	SA-1	SA-3	BC-3	DSG	DP
O-1	0.44	0.40	0.27	0.65	0.39	0.91	0.10	0.48	0.15	0.04	0.00
O-2	0.44	0.40	0.27	0.65	0.39	0.91	0.10	0.48	0.15	0.04	0.00
O-3	0.86	0.90	0.20	0.82	0.18	0.34	0.94	0.00	0.03	0.17	0.00
O-4	0.48	0.12	0.43	0.98	0.05	0.35	0.98	0.66	0.25	0.04	0.00
O-5	0.16	0.02	0.17	0.05	0.02	0.82	0.49	0.30	0.16	0.21	0.00
O-6	0.01	0.01	0.19	0.63	0.12	0.50	0.39	0.58	0.98	0.78	0.08
O-7	0.16	0.01	0.12	0.15	0.14	0.06	0.29	0.55	0.07	0.33	0.00
O-8	0.05	0.18	0.38	0.84	0.06	0.00	0.01	0.01	0.31	0.27	0.85
O-9	0.34	0.08	0.04	0.58	0.69	0.05	0.53	0.00	0.23	0.52	0.03
O-10	0.34	0.08	0.04	0.58	0.69	0.05	0.53	0.00	0.23	0.52	0.03

Task 2 – VVER-1000





Scenarios, Scope, SA Tools, Methodologies

Institution	Scenario	SA code	Framework of analysis
IBRAE (Russian Federation)	**LBLOCA with HPI and LPI failure with unmitigated SBO (In-vessel)**	SOCRAT	Regulation (NP-001-15)
ENERGOATOM (Ukraine)	**SBO (In-vessel)**	MELCOR	Requirements from IAEA and European manufactures
KURCHATOV (Russian Federation)	**LB LOCA + SBO including (In-vessel)**	SOCRAT	Regulation (NP-001-15)
GIDROPRESS (Russian Federation)	**Unmitigated SBO (In-vessel)**	SOCRAT	Regulation (NP-001-15)

Institution	Sampling and UQ method	UQ tool and calculation scheme	Sensitivity/importance analysis
IBRAE	SRS (ASME V&V 20) **N = 200**	In-house	Pearson, Spearman
ENERGOATOM	LHS **N = 400**	In-house	Pearson, Spearman
KURCHATOV	SRS **N = 275**	In-house	Spearman, Kendall
GIDROPRESS	SRS (two-sided Wilks, 95%/95%)	In-house	Pearson, Spearman,Kendall

UP and FOMs

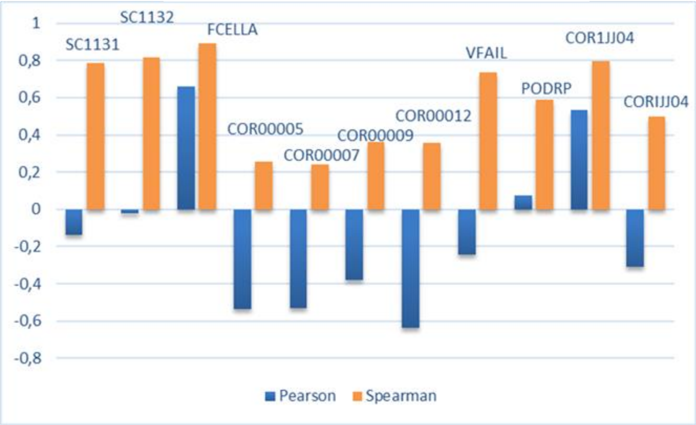
Institution	UP	UP's Domain
IBRAE (Russian Federation)	23	boundary conditions, geometric parameters and modelling parameters that affect corium behavior at reactor bottom
ENERGOATOM (Ukraine)	19	models of fuel and melts behavior
KURCHATOV (Russian Federation)	134	core discretization
GIDROPRESS (Russian Federation)	42	geometrical parameters, initial and boundary conditions, thermophysical properties of materials and setpoints of equipment

Institution	FOM
IBRAE (Russian Federation)	**Total H2 production in the core during the accident**
ENERGOATOM (Ukraine)	**Total H2 production in the core during the accident**
KURCHATOV (Russian Federation)	**Timings of core degradation events**, mass of H2 and containment atmosphere parameters
GIDROPRESS (Russian Federation)	**1) Timings of core degradation events** 2) parameters of the **melt released** from the reactor vessel 3) maximal values of **pressure in the primary, secondary, and containment** 4) mass of H2 and containment atmosphere parameters

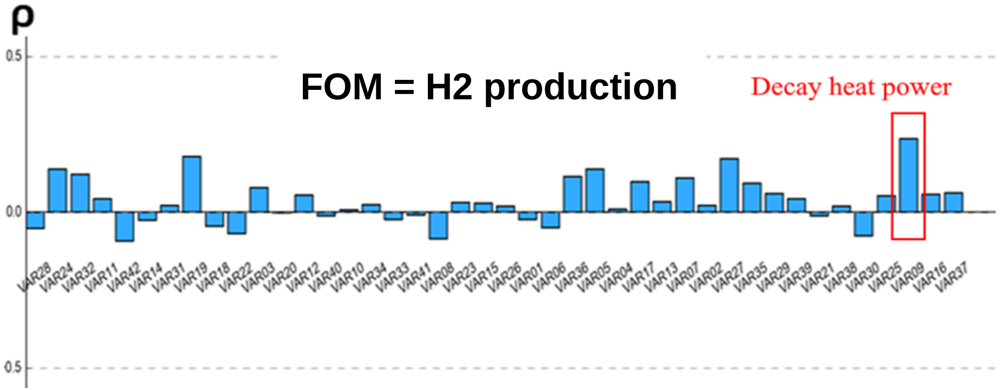
Results (1/2)

- **ENERGOATOM**
  - **Vessel remain intact in 208 calcs.**
  - **Failure of the reactor vessel in 126 calcs.**
  - **66 calcs. excluded (lack of convergence)**

FOM = H2 production



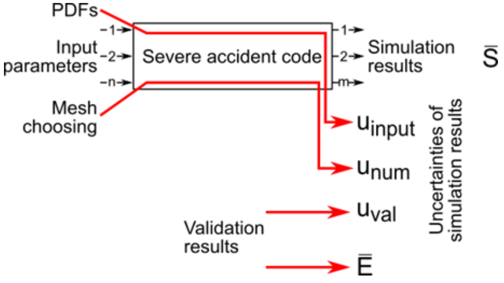
- **GIDROPRESS**



FOM = H2 production

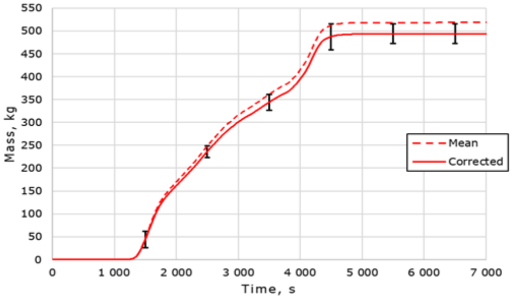
Results (2/2)

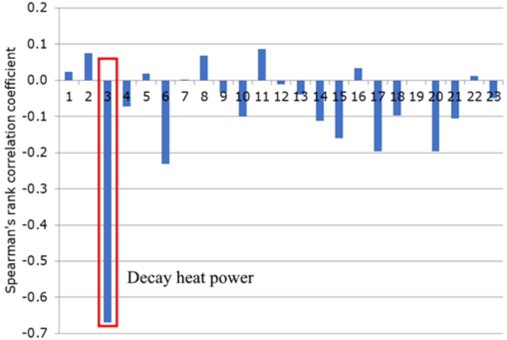
- **IBRAE RAN**
- Uncertainty analysis results of SA should be corrected taking into account code validation results



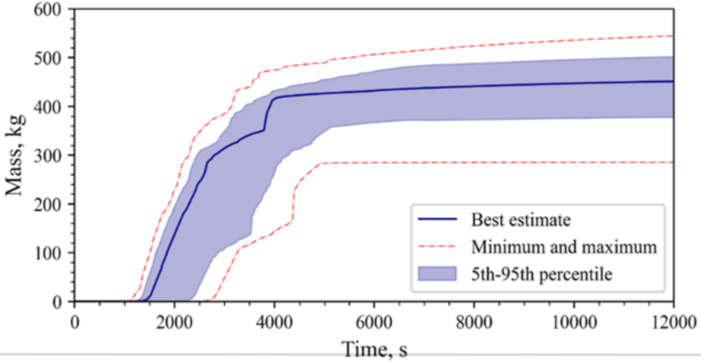
- **KURCHATOV INSTITUTE**

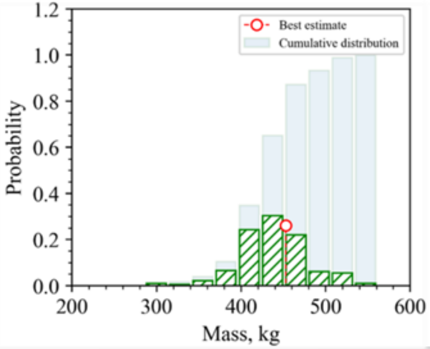
FOM = H2 production



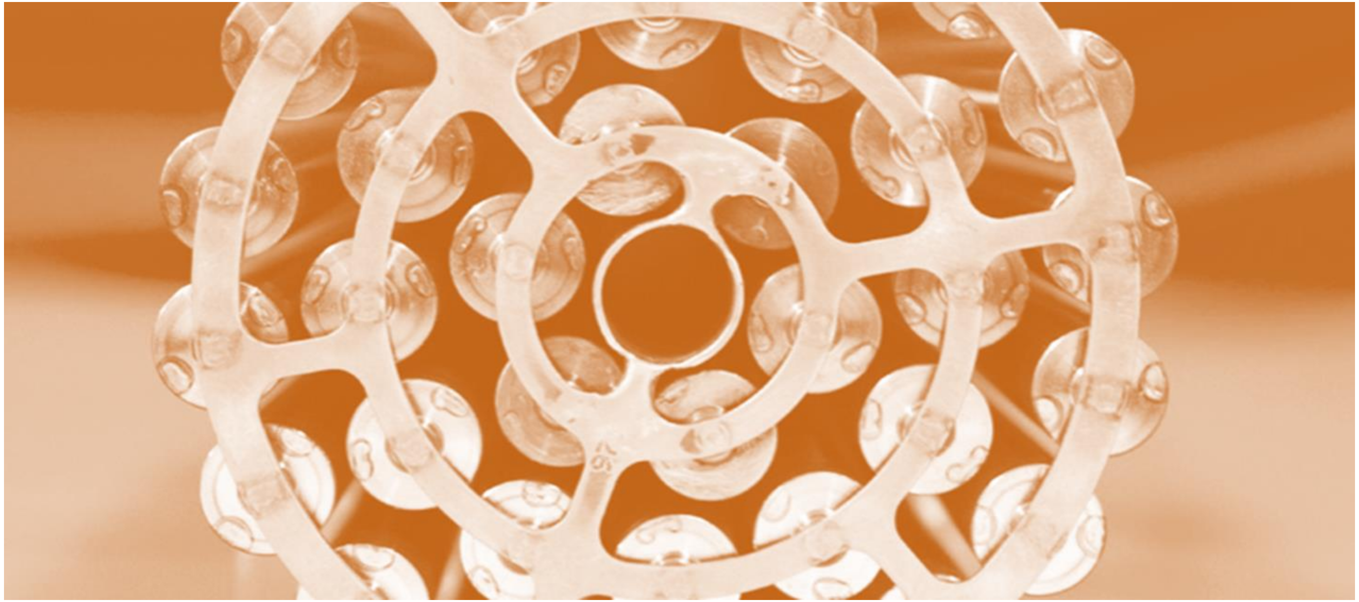


FOM = H2 production





Task 2 – CANDU 6



Scenarios, Scope, and SA/U&SA Tools

Institution	Scenario	SA code	Framework of analysis
KAERI (Rep. Of Korea)	**SBO** **(In-vessel)**	CAISER	Support to SAMG development
UPB (Romania)		RELAP/SCDAPSIM/MOD3.6	
CNL (Canada)		MAAP-CANDU v5.00A	

Institution	Sampling and UQ method	UQ tool and calculation scheme	Sensitivity/importance analysis
KAERI (Rep. Of Korea)	Monte-Carlo method Response Correlation/Regression Analysis Methodology	MOSAIQUE	Pearson, Spearman
UPB (Romania)	GRS method (Wilks approach)	Integrated Uncertainty Analysis package in RELAP/SCDAP-SIM	Pearson, Spearman
CNL (Canada)	GRS method (Wilks approach)	SUSA	Pearson, Spearman

UP and FOMs

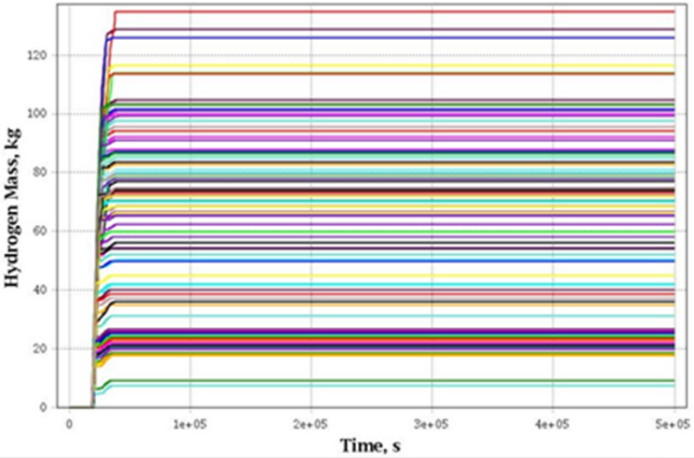
Institution	UP	UP’s Domain
KAERI (Rep. Of Korea)	13	Core heat-up Debris formation Core collapse Fission Product releases
UPB (Romania)	26	
CNL (Canada)	26	

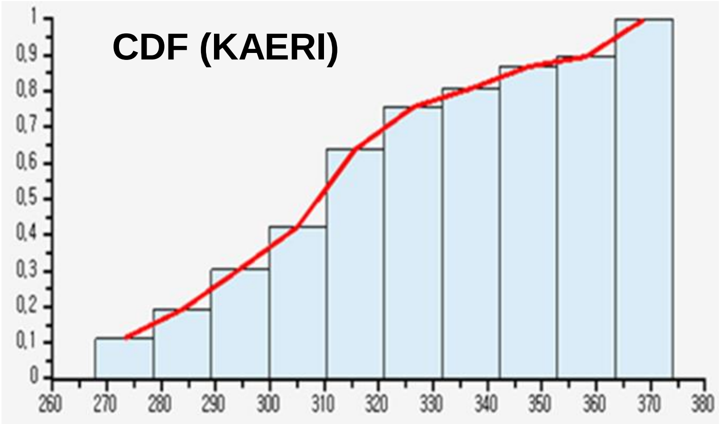
Institution	FOM
KAERI (Rep. Of Korea)	**In-/ex-vessel hydrogen generation** **Fission product mass released to the environment** **Event timings such as calandria vessel failure time**
UPB (Romania)	
CNL (Canada)	

Results (1/2)

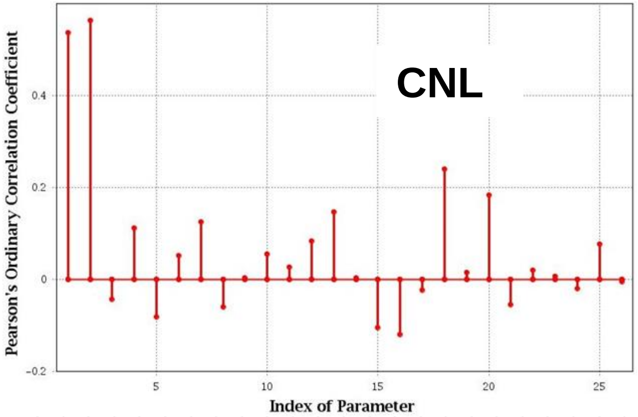
➢ H2 In-vessel Production

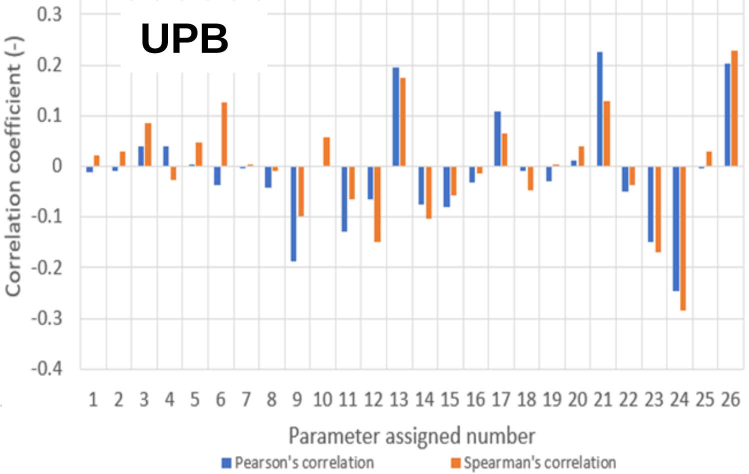
Institution	Lower tolerance limit, kg	Upper tolerance limit, kg	Comments
KAERI	15.0 (7.5 per PHTS loop)	269.4 (134.7 per PHTS loop)	Two-sided, 95%/95%
	N/A	258.0	One-sided, 95%/95%
UPB	273.3	370.6	Two-sided, 95%/95%
CNL	Lower tolerance limit, kg	Upper tolerance limit, kg	Comments





Results (2/2)





es in

Institution	The most significant parameter	The 2nd most significant parameter
CNL	MLOAD(1): the maximum allowable amount of suspended core debris in a loop	VFSEP: the maximum PHTS void fraction where the primary coolant is modelled as a homogeneous two-phase mixture
UPB	contact angle for pressure tube to calandria tube	the fuel channel rupture area
KAERI	Vfactor_PT: the view factor between a fuel rods and a pressure tube	pt_local a fuel channel failure temperature caused by a local melting of a pressure tube

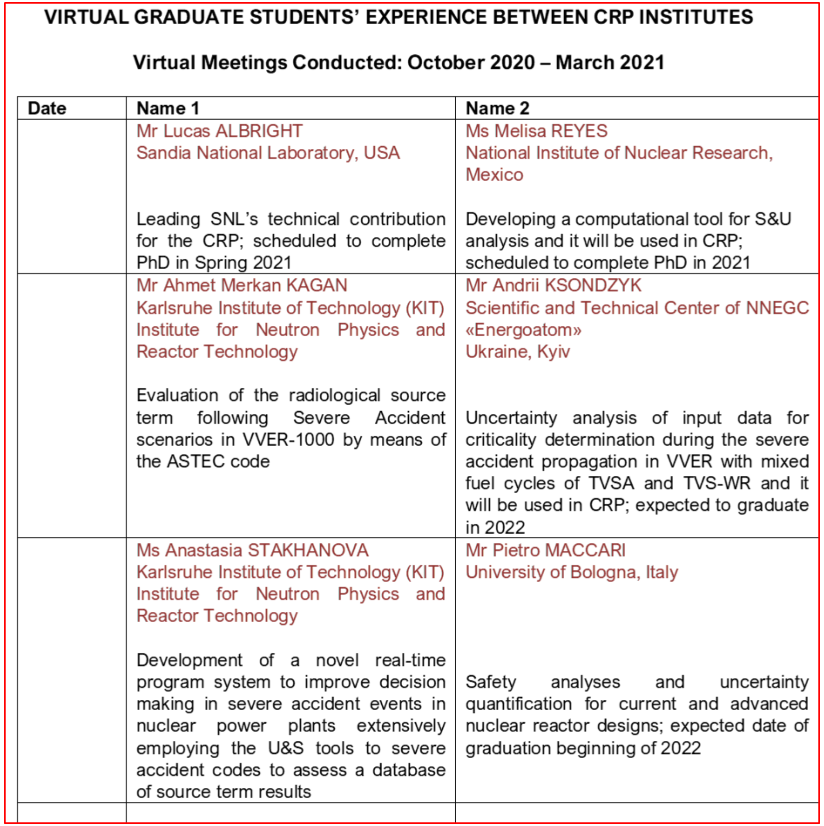
Dissemitation and Education

Source: https://allchildrenlearning.org/assessment-topics/using/reporting-and-dissemination/

Outputs

- **Five IAEA TECDOCs (thanks Tatjana☺!)** on uncertainty methods and tools for SA codes with relevant benchmark results → **on-going**
- **IAEA NES** on state of practice with lessons learned on best practices in U&S methodologies for the severe accidents analyses in WCRs → **2024**
- Relevant **topical meetings** organized → **on-going**
- Publications in conference proceedings and peer reviewed journals, i.e. 2022 IAEA TIC, ERMSAR2022 → **done/on-going**
- **Graduate Students Programme** to strengthen promotion of research on SA simulation and modelling in developing Member States through pair building between agreement holders and contract holders institutes → **done/on-going**

Graduate Students Programme



VIRTUAL GRADUATE STUDENTS' EXPERIENCE BETWEEN CRP INSTITUTES

Virtual Meetings Conducted: October 2020 – March 2021

Date	Name 1	Name 2
	Mr Lucas ALBRIGHT Sandia National Laboratory, USA Leading SNL's technical contribution for the CRP; scheduled to complete PhD in Spring 2021	Ms Melisa REYES National Institute of Nuclear Research, Mexico Developing a computational tool for S&U analysis and it will be used in CRP; scheduled to complete PhD in 2021
	Mr Ahmet Merkan KAGAN Karlsruhe Institute of Technology (KIT) Institute for Neutron Physics and Reactor Technology Evaluation of the radiological source term following Severe Accident scenarios in VVER-1000 by means of the ASTEC code	Mr Andrii KSONDZYK Scientific and Technical Center of NNEGC «Energoatom» Ukraine, Kyiv Uncertainty analysis of input data for criticality determination during the severe accident propagation in VVER with mixed fuel cycles of TVSA and TVS-WR and it will be used in CRP; expected to graduate in 2022
	Ms Anastasia STAKHANOVA Karlsruhe Institute of Technology (KIT) Institute for Neutron Physics and Reactor Technology Development of a novel real-time program system to improve decision making in severe accident events in nuclear power plants extensively employing the U&S tools to severe accident codes to assess a database of source term results	Mr Pietro MACCARI University of Bologna, Italy Safety analyses and uncertainty quantification for current and advanced nuclear reactor designs; expected date of graduation beginning of 2022

Based on the graduate students' pair building between CRP institutes

Graduate students are asked to select one day in a semester to spend time together virtually, and to prepare a brief report to IAEA

Their virtual experience should include the following:

- Virtual breakfast to get to know each other
- Present with well-developed slides the graduate (PhD or MS) research and how it relates to the CRP objectives and scopes
- Spend virtual afternoon working on a common topic
- Write jointly the report and submit to IAEA Chairs

Conclusions

- **Several and extensive SA/UT successfully applied to different nuclear technologies**
- **Large amount of UPs identified and characterized in terms of PDFs**
- Additional efforts needed on:
  - The proper selection of the UP and the assessment of the corresponding PDFs
  - The definition of the number of code calculations → much larger than the number of UPs to limit the impact of random code failures and then to guarantee reliable sensitivity measures
  - The use of different correlations and/or multiple linear regression-based sensitivity/importance measures
- **5 TECDOCS and a NES in preparation**
- **Topical meetings and Graduate Students Programme** assessed to further sharing the knowledge gained among the Member States

Conclusions and Outlook

- **Solid basis of understanding on the capabilities of the SA/UT platforms assessed**
- Huge experience on the performance of SA and SA/UT gained on the application to different technologies on
  - Scenarios
  - Accident phases
- **Database of SA analyses assessed by the different Member States joining the CRP as training data for the development of surrogate models**

Acknowledgement

Thanks a lot to all the members of the CRP for having provided a summary of the work performed in each task.

A special thank to T. Jevremovic, IAEA chair of the CRP.