FASTNET SCENARIOS DATABASE DEVELOPMENT AND STATUS

F. Mascari[1], F. Rocchi[2] P. Carny[2], L. Liptak[2], M. Adorni[3], J. Fontanet[4], L.E. Herranz[4], M. Shawkat[5], W. Raskob[6], F. Cousin[7], J. C. de la Rosa Blul[8], E. Urbonavicius[9], F. Di Dedda[10], T. Augustsson[11], M. Constantin[12], G. Arbaev[13], P. Isaksson[14], J. Kubicek[15]

[1]: ENEA, via Martiri di Monte Sole 4, 40129, Bologna, Italy
[2]:ABmerit, s.r.o., Hornopotocna 1, P.O.Box 200, 917 01 Trnava, Slovakia
[3]: BelV, Rue Walcourt 148, B-1070 Brussels, Belgium
[4]: CIEMAT, Av. Complutense 40, 28040 Madrid, Spain
[5]: CNSC, 280 Slater Street, P.O. Box 1046, Station B, Ottawa, ON, K1P 5S9, Canada
[6]: KIT, Hermann-von-Helmholtz Platz 1, 76344 Eggenstein-Leopoldshafen, Germany,
[7] IRSN, C.E. Cadarache, BP3 13115 Saint-Paul-lez-Durance Cedex, France
[8]: European Commission JRC, Westerduinweg 3,
1755 LE Petten, The Netherlands
[9]: LEI, Breslaujos g. 3, LT-44403 Kaunas, Lithuania
[10]: LR, Göteborgsvägen 74, 433 63 Sävedalen, Sweden
[11]: OKG, OKG Aktiebolag, Simpevarp 200, 572 83 Oskarshamn, Sweden
[12]: RATEN ICN, 1, Campului St., 115400-Mioveni, AG, Romania.
[13]: SEC NRS, Russia, 107140, Moscow, Malaya Krasnoselskaya st. 2/8, bld. 5
[14]: SSM, SSM: Strålsäkerhetsmyndigheten, 171 16 Stockholm,Sweden
[15]: UJV Rez a. s., Hlavni 130, Husinec - Rez, 250 68, Czech Republic

fulvio.mascari@enea.it
federico.rocchi@enea.it
carny@abmerit.sk
liptak@abmerit.sk
martina.adorni@Belv.be
joan.fontanet@ciemat.es
luisen.herranz@ciemat.es
mohamed.shawkat@canada.ca
wolfgang.raskob@kit.edu
frederic.cousin@irsn.fr
juan-carlos.de-la-rosa-blul@ec.europa.eu
egidijus.urbonavicius@lei.lt
Francesco.DiDedda@lr.org
tomas.augustsson@okg.uniper.energy
marin.constantin@nuclear.ro
arbaev@secnrs.ru
Patrick.Isaksson@ssm.se
jan.kubicek@ujv.cz

ABSTRACT

In the Horizon 2020 Framework Programme of the European Commission, the FASTNET project – Fast Nuclear Emergency Tools – has been funded and will finish to September 2019. This project, coordinated by Institut de Radioprotection et de Sûreté Nucléaire (IRSN), has several objectives, among them the set-up of a Severe Accident (SA) source term database to develop and qualify a common graded response methodology for Emergency Preparedness and Response (EP&R). This methodology should integrate several tools (e.g. fast running nuclear emergency tools) and methods for both diagnosis and prognosis of SA progression and the related estimation of consequences, in terms of source term release, for EP&R Centers. The predicted source term is subsequently used as an input for decision support systems estimating the consequences to the public and the identification of the population protection measures. The FASTNET project includes participants from EU, Canada, USA.

In this respect, the development of a common database of pre-calculated SA scenarios, trying to cover all concepts of existing plants in Europe and participating countries is one of the key activities of the project. Considering the complexity and mutual interacting and interrelated phenomena/processes along a SA transient progression, the most accurate simulation tools to estimate the related source terms are the State-of-Art SA integral codes (e.g. ASTEC, MAAP, MAAP-CANDU, MELCOR, etc.). These codes have been used to develop the source term database of the project and many other relevant output variables necessary to characterize the SA transient progression. Based on the work done by the FASTNET Senior Expert Group, a set of representative scenarios (LBLOCA, SBO, etc.) have been suggested to the project partners trying to cover a wide range of scenarios of generic plant designs found in Europe and other participating countries (e.g. BWR Mark-I like, CANDU like, PWR 900 and 1300 like, VVER 400 and 1000 like).

In order to get the source term release, all the different phases of SA scenarios progression have been simulated and, to facilitate the comparison between different reactor designs, a similar minimum set of Figures Of Merit (FOM) have been agreed by the project partners and investigated in each SA code application. The goal of this paper is to discuss the criteria adopted to pick the FOMs, to summarize the SA-code applications, and to discuss the applicability of this database for supporting the development of fast nuclear emergency tools.

KEYWORDS
Severe Accident, source term release, Emergency Preparedness and Response, FASTNET, decision support

1. INTRODUCTION

After the Fukushima accident in March 2011, the worldwide interest has been focused on Severe Accident (SA) [1-3] management strategy improvement and on the development of fast running tools to estimate the potential source term release in a postulated Severe Accident (SA) [4]. A SA is a Beyond Design Basis Accident (BDBA) involving significant core degradation. If the containment fails, due to the postulated initial and extreme boundary conditions, a source term release to the environment can occur.

The Fukushima accident did not only encourage the development of these fast running tools, but also highlighted the need for further harmonization efforts at European level regarding what should be recommended to the general public during an emergency situation [4]. Therefore, the need of common and consolidated methodologies was raised as a key need. These methodologies should involve experts and computational tool applications aimed at the estimation of the consequence of an accidental release to the environment and its impact on the population.

In this framework, the FASTNET – Fast Nuclear Emergency Tool – project, funded from the H2020 Framework Programme of the European Commission and coordinated by IRSN, has the main objective to develop and validate a common graded response methodology that integrates several tools and methods to be used for potential SA progression analyses and consequence estimations [5]. The main aim of this response methodology, to be applied during a postulated accident, is to characterize the actual status of a plant during the transient evolution (diagnosis of the plant status) and estimate the potential evolution of the severe accident scenarios (prognosis). Considering the concurrent malfunctions and extreme boundary conditions that could occur during a SA evolution, the estimation and evaluation of a potential source term release to the environment is one of the key elements of this methodology. Another goal of the FASTNET project is to establish the connection between the FASTNET tools and other systems that use these source terms for further consequence assessments. As shown in Fig. 1, the project is divided in 6 Work Packages (WP) in order to achieve these goals.

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

Figure 1: FASTNET project WPs [5].

The development of a common database of pre-calculated SA scenarios evolution, trying to cover a wide range of scenarios of plant designs found in Europe and other participating countries (e.g. Canada) and object of the WP1, coordinated by LEI, is one of the key activities of the project. State-of- the- Art SA integral codes (e.g. ASTEC [6], MAAP [7], MAAP-CANDU [8], MELCOR [9], etc.) have been used to develop the source term database of the project and many other relevant output variables necessary to characterize the SA transient progression.

The fast running tools identified at European Level to be part of the FASTNET methodology are the deterministic code PERSAN [10], developed by IRSN (France), and the probabilistic code RASTEP [11], developed by Lloyd's Register (Sweden), based on Bayesian Belief Networks (BBNs) and pre-calculated source terms. Since one of the main targets of the SA database is to be used for the validation of these tools a list of variables has been selected having as a reference [12-14] the needs of PERSAN and RASTEP.

The main target of this paper is to give a general overview of the scenarios database, its structure and the current status. Though the SA codes provide State-of-the-Art predictions of a potential source term, it is necessary to stress that the results are affected by uncertainties related to models/correlations and user input. For this reason, the final section of the paper briefly describes the general source of uncertainties in SA codes.

2. SCENARIOS DATABASE CONCEPT OVERVIEW

In order to support the development of the selected fast running tools, a suitable reference database has been developed with the State-of-the-Art SA code. This database can be used to benchmark and analyze the capability of the fast running codes to predict source terms related to postulated SA sequences. Having in mind the need of emergency preparedness, and based on the work done by the FASTNET Senior Expert Group setup in the project, a set of most representative scenarios (LBLOCA, SBO, etc..) has been suggested to the project partners trying to cover wide range of scenarios of generic plant designs found in Europe and other participating countries (e.g. BWR Mark-I like, CANDU like, PWR 900 and 1300 like, VVER 400 and 1000 like, etc). Along the project the list developed by the Senior Expert Group has been reviewed by project partners and a final list of representative scenarios has been defined to be included in the database.

The institutions involved in the database development are ABMERIT, BOKU, CIEMAT, CNSC, ENEA, IRSN, JRC, LRC, NRI, RATEN, SECNRS. The SA code used for the development of the database are ASTEC [6], MAAP[7] and MAAP-CANDU[8], MELCOR [9]. Table I shows the scenarios simulated for each generic design, Table II shows the scenario matrix covered by the FASTNET database. At this stage attention was paid to full power operation only (no low power and shut down states scenarios are included in the database).

Two main goals were addressed by this database. The first is related to test the methods for emergency response, and the second is to support the further development of the tools. For the first objective and since during emergency situation the number of data coming from the affected plant may be limited, a first reduced list of variables has been selected. These parameters are mostly useful to characterize the status of the plant and its safety during a postulated emergency situation and can be used to assess the EP&R methodology. Secondly, to improve the calculation tools, an extended list of parameters / data is defined to get a more detailed understanding of the scenario accident progression and how the fast running code is able to reproduce it. Considering these two main objectives the data were divided into three mains types: a) Data needed for emergency response (this is a reduced set of data); b) Data needed to validate fast running tools (this set of data also includes the data in a)); c) A global set of data to define accident scenario progress.

Table I: Scenarios simulated for each generic design

Reference Generic Reactor	Scenarios
VVER440	Large LOCA on hot leg (full guillotine rupture on hot leg) PRISE, break 13.5 cm2 on hot collector of SG No.1 (no retention of fission products in tubes of SG) Small PRISE, break size 2.7 cm^2, feedwater unavailable SBO with in-vessel retention (IVR) procedure applied (cavity flooded)

VVER 1000	SBLOCA SBO with containment failure SBO with containment intact
BWR MARK-I	LOCA SBO at high pressure SBO at low pressure
Generic CANDU design	SBO (1 unit in a single unit containment) SBO (4 units in a shared containment) SBLOCA (1 unit in a single unit containment) LBLOCA (1 unit in a single unit containment) SGTR (1 unit in a single unit containment) SGTR (1 unit in a shared containment)
PWR 900MWe	SBO SBO-Early Failure
French REP 1300 P'4	LOCA 2P, no SI, no CSS LOCA 2P, no SI, CSS lost in post recirculation LOCA 2P, no SI, CSS lost at recirculation LOCA 2P, no CSS, SI long term LOCA 2P, CSS lost after recirculation, SI long term LOCA 2P, CSS lost at recirculation, SI long term LOCA 2P, CSS long term, SI lost after recirculation LOCA 2P, CSS lost after recirculation, SI lost after recirculation LOCA 2P, no CSS, SI lost at recirculation LOCA 2P, CSS lost after recirculation, SI lost at recirculation LOCA 2P, CSS lost at recirculation, SI lost at recirculation LOCA 6P, no SI, CSS lost in post recirculation LOCA 6P, no SI, CSS lost at recirculation LOCA 6P, no CSS, SI long term LOCA 6P, CSS lost after recirculation, SI long term LOCA 6P, CSS lost at recirculation, SI long term LOCA 6P, CSS long term, SI lost after recirculation LOCA 6P, CSS lost after recirculation, SI lost after recirculation LOCA 6P, no CSS, SI lost at recirculation LOCA 6P, CSS lost after recirculation, SI lost at recirculation LOCA 6P, CSS lost at recirculation, SI lost at recirculation LOCA 12P, no SI, no CSS LOCA 12P, no SI, CSS lost at recirculation LOCA 12P, SI lost at recirc, CSS lost after recirculation LOCA 12P, SI lost at recirc, CSS lost at recirculation LOCA 12P, HL no SI, CSS lost in post recirculation LOCA 12P, HL no SI, no CSS
Westinghouse PWR (Doel 4 LIKE)	12"Cold Leg ISLOCA with all safety systems available, 4"containment isolation failure 12"Cold Leg ISLOCA with all safety systems available, FCVS available 12"Cold Leg ISLOCA with all safety systems available, FCVS unavailable 12"Cold Leg ISLOCA with all safety systems available, Containment failure when RPV failure 2"Cold Leg SBLOCA with recirculation failure and CSS available, 4"containment isolation failure 2"Cold Leg SBLOCA with recirculation failure and CSS available, FCVS available 2"Cold Leg SBLOCA with recirculation failure and CSS available, FCVS

	unavailable 2"Cold Leg SBLOCA with recirculation failure and CSS available, Containment failure when RPV failure 12" Cold Leg LBLOCA, ECCS unavailable and CSS available, 4"containment isolation failure 12" Cold Leg LBLOCA, ECCS unavailable and CSS available, FCVS available 12" Cold Leg LBLOCA, ECCS unavailable and CSS available, FCVS unavailable 12" Cold Leg LBLOCA, ECCS unavailable and CSS available, Containment failure when RPV failure SBO, AFW unavailable, 4"containment isolation failure SBO, AFW unavailable, FCVS available SBO, AFW unavailable, FCVS unavailable SBO, AFW unavailable, Containment failure when RPV failure Loss-of-cooling in the Spent Fuel Pool Large loss-of-coolant in the Spent Fuel Pool
ABB_BWR	LOCA, no ECCS, FCVS, no CSS. Transient, no ECCS, FCVS, CSS. LOCA, no ECCS available, no FCVS, CSS. ATWS, FCVS, subsequent LOCA. Transient, no RHR, ECCS, FCVS, no CSS. Transient, no ECCS, no FCVS, CSS. ATWS, no ECCS, FCVS, subsequent LOCA.

Table II: Scenarios matrix considered for each generic design

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

Since each scenario has to be described in a unified way to be included in the database, a special spreadsheet format has been used to collect the results. This is based on a list of the key parameters for description of the accident scenario that was agreed on by all the partners. In order to characterize the transient evolution several selected parameters related to their behavior versus time have been provided and several time-value tables were provided as well. The data has been grouped as:

- Basic data regarding the plant:
  A minimum set of parameters to identify a generic plant, used to have a common definition of the plant. Table III shows the generic plant data considered for the construction of the database.

- Initial Inventory:
  The initial inventory is a necessary input for using the fast running tools, therefore this is one of the input-deck characterization requested to each partner. Considering the different needs, the initial inventory of the reactor has been set by the code user.

- Scenarios description:
  In order to describe the evolution of the scenarios, a detailed list of parameters was selected considering the different management actions and the safety systems availabilities and actuation (manual or automatic). In this sense, for each initiator and plant, a table of chosen events was established. By default, the user has to combine each events (YES/NO) to build the scenario. Table IV shows the reference table to be filled in by WP1 partners for each scenario.

- Key events:
  The key events table, including the relevant phenomenological aspects of a severe accident evolution, was used to characterize sequence of main events of the transients. Examples of key parameters collected by WP1 partners are the "Start of core uncovering", the "Time of lower head failure" and the "Time of large scale relocation of core debris to lower plenum" which are useful parameters to be able to characterize the core degradation time history. This event timing is complementary with the scenario description parameter list described above. Table V shows the complete list of parameters requested to characterize the key event of the transient scenarios.

- Physical data regarding core behavior
  The core degradation evolution determines the radionuclide release during the SA scenario. Therefore, its characterization is fundamental in the development of a source term database development. Examples of key parameters vs time evolution, collected by WP1 partners, are the "Core exit temperature" useful to characterize the core degradation, the "Core water level" useful to characterize the eventual reflooding due to the ECCS, "DTsat in core" useful to characterize the vaporization in the core, "H2 mass generated in core" useful to characterize the core degradation evolution. Table VI shows the complete list of parameters requested to characterize the physical behavior of the core.

- Physical data regarding primary circuit behavior
  The primary system thermal hydraulic behavior has a key influence on the core degradation. The thermal hydraulic transient scenarios that can take place considering the different mitigation actions postulated in the Table IV, influences the core degradation behavior and the consequent radionuclide release. Examples of key parameters collected by WP1 partners are the "Primary circuit pressure", the "Emergency core injection flow rate", the "Pressurizer Safety/Relief valve operation/flow rate", the "Mass flow rate at RCS break" (break size), the "Charging/let down line mass flow rate", the "FP retention in RCS". Table VII shows the complete list of parameters requested for the primary circuit behavior.

- Physical data regarding secondary circuit behavior:
  The characterization of the secondary side is important as well because it can drive the primary pressure evolution, and, on the other side, it should characterize also the potential secondary side contamination. Examples of key parameters collected by WP1 partners are the "Secondary Pressure" useful to characterize the effect of the secondary side on the primary side, the "SG isolation (water and vapour)" useful to characterize the secondary side contamination, the

"Atmospheric valve mass flow rate" useful to characterize the direct release to the environment, the "Activity in SG" useful to characterize SGTR transient etc. Table VIII shows the complete list of parameters requested to characterize the physical behavior for the secondary side behavior.

- <u>Physical data regarding containment behavior</u>
  Considering the key role of the containment as a last barrier, a detailed characterization of the containment has been requested to the partners in order to define the status and the effect of the mitigation actions. Examples of key parameters collected by WP1 partners are the "Containment Pressure", useful to characterize all types of transients, "Total Flow rate of containment spray system" useful to characterize the containment depressurization and FP deposition, the "In-vessel hydrogen generation rate" and "Ex-vessel hydrogen generation rate" useful to characterize the hydrogen production, the "Mass of molten material in reactor pressure vessel pedestal and drywell", etc. Table IX shows the complete list of parameters requested for the containment behavior.

- <u>Physical data regarding release</u>
  Since the main target of WP1 is to develop a source term database a detailed characterization of the possible leak paths, direct releases from the different buildings and FP release/transport/deposition phenomena is necessary to quantify the release from the plant. Examples of key parameters collected by WP1 partners are the "Leak of containment building to each building", the "Leak of the auxiliaries buildings", the "Volumetric activity in chimney" (noble gases, volatile element, semi-volatile elements, molecular iodine, organic iodine, (non volatile) aerosols, isotope), the "FP concentration evolution in reactor building for each FP family and isotope". Table X shows the complete list of parameters requested to characterize the release.

- <u>Suggested element and isotopes to calculate release</u>
  An exhaustive list of 147 isotopes also including different iodine compounds (i.e., molecular iodine, organics, oxides, etc.) has been defined. This list must be understood as a 'wish list' since, due to specific characteristics of each SA accident codes used to simulate the different scenarios, the final achievable list is necessarily shorter. Some of the suggested elements and isotopes to calculate the release from the fuel and the source term to the environment are presented in Table XI. An extended list to help emergency development tools such as PERSAN is reported in Table XII.

- <u>Other data requested</u>
  Other data have been requested to the WP1 participants to characterize:
  - Iodine chemistry, as shown in Table XIII;
  - CANDU, as shown in Table XIV;
  - SFP, as shown in Table XV;
  - Other.

Table III: Basic data regarding the generic plant considered

Generic power plant type	**VVER 440**	**VVER1000**	**BWR**	**BWR**	**CANDU**	**PWR**	**French REP 1300 P'4**	**PWR**
Coolant type	Light water	Light water	Light water	Light water	Heavy water	Light water	Light water	Light water
Moderator type	Light water	Light water	Light water	Light water	Heavy water	Light water	Light water	Light water
Fuel type	UO2 enrichment : 4.38% U235	UO2 - 3.3% U-235 enrichment	UO2 (2.7% U-235)	UO2	UO2 (natural uranium of 0.72wt% of U-235)	UO2 enrichment	UO2	UO2 enrichment
Mass of U (kg)	40200	78924.2	70000	76000	87800 (single unit containment case) 125000 (single unit in shared containment)	71200	103725	85529
Average burnup (MWd/kgU)	44	33	21.5	46	8.0 (single unit containment case) 8.4 (single unit in shared containment)		2000MWD / T	26.053
Containment type	Bubbler condenser (confinement)	Dry containment	Mark-I	Generic ABB BWR	*	Dry containment	Volume = 77 000 m3	Dry large double containment
Internal or external recirculation loop		N/A	External	External	2180 (single unit containment case) 2680 (single unit in shared containment)	External	External	External
Design thermal power (MW)	1444	3000	1400	1800		2775	3800	3000
Design electrical power (MW)	500	1000	470	650	Assuming 30% overall efficiency 654 (single unit containment case) 804 (single unit in shared containment)	900	1300	1000

* For single unit containment it is a dry containment maintained slightly sub-atmospheric with dousing tank at the top of the containment. For shared containment, a dry containment with a vacuum building that has a dousing tank is connected to the containment in case of accident. Containment is shared between 4 units and has and Air Cooling Units, H2 igniters, and re-combiners.

Table IV: Scenarios description table

Availability of offsite power	Yes/No
Availability of ultimate heat sink	Yes/No
ECCS (Emergency Core Cooling System) availability	Yes/No
ECCS automatic/manual activation (HPCI/LPCI according to plant design and need)	Yes/No
LPI set point	Yes/No
RHR (Residual Heat Removal) availability	Yes/No
RHR automatic/manual activation	Yes/No
Primary pump availability (Recirculation loop flow)	Yes/No
Recirculation systems availability,	Yes/No
Feedwater availability	Yes/No
RCIC (Reactor Core Isolation Cooling) availability	Yes/No
RCIC/AFW automatic/manual activation	Yes/No
ADS (Automatic Depressurisation System) Availability	Yes/No
ADS automatic/manual activation	Yes/No
SRV automatic/manual activation	Yes/No
SRV stuck open excessive cycling	Yes/No
Standby Liquid Control System (SLCS) availability	Yes/No
Boron injection system availability	Yes/No
SLCS automatic/manual activation	Yes/No
Boron automatic/manual activation	Yes/No
IC (Isolation Condenser) availability	Yes/No
IC automatic/manual activation	Yes/No
Feed water availability (tank level)	Yes/No
Containment status (1. OK, 2. Failure, 3. Non-Isolated, 4. Bypass, 5. I-LOCA)	Yes/No
Hydrogen recombiners/ignitors availability	Yes/No
Containment spray availability (drywell/wetwell)	Yes/No
Containment spray automatic/manual activation	Yes/No
Containment depressurization system availability	Yes/No
Containment depressurization system automatic/manual activation	Yes/No
Lower drywell filling system availability	Yes/No
Status of bubble tower (VVER-440 only: drained, not drained)	Yes/No
Ex-vessel cooling (IVR-ok, IVR without circulating cooling water, no IVR)	Yes/No
Containment isolation	Yes/No
Filtered containment venting system availability	Yes/No
Filtered containment venting system automatic/manual activation	Yes/No
Lower drywell filling system availability	Yes/No
Lower drywell filling system activation	Yes/No
Battery back-up availability	Yes/No
Diesel back-up availability	Yes/No

Turbine condenser availability	Yes/No
Scrubber systems availability in auxiliary building (VVER1000)	Yes/No
Nominal Leak rate of containment building to annulus space at 5 bars	
Nominal Leak rate of containment building to fuel building at 5 bars	
Nominal Leak rate of containment building to nuclear auxiliary building at 5 bars	
Nominal Leak rate of containment building to safety auxiliary building at 5 bars	
Nominal Leak rate of safety auxiliary building to environment	
Nominal Leak rate of nuclear auxiliary building to environment	
Nominal Leak rate of fuel building to environment	
Nominal Leak rate of containment building to environment at 5 bars	
Nominal Leak rate of containment building to environment (filtered)	

Table V: Sequence of the main event of the transient

Event	Unit
Time of SCRAM	s
Time of fuel channel dryout	s
Time of coolant flashing in BWR core	s
Time of SG loop dryout	s
Time of rupture disks burst for CANDU	s
Calandria vessel rupture time	s
Calandria water boil-off time	s
Time when water in SFP starts boiling	s
Start uncovering the core	s
Time of total core uncover	s
Time of large scale relocation of core debris to lower plenum	s
Time of lower head vessel failure	s
Core collapse time	s
Time of containment failure	s
Time of RPV rupture	s
Time of basemat meltthrough	s
Opening time of the containment depressurization system	s
Time of the raft breakthrough	s

Table VI: Physical data regarding core behavior

Parameter	Unit
Core exit temperature	K
Core water level	m
Average core temperature	K
Peak core temperature	K

Degraded mass, i.e. molten + debris + relocated	kg
Corium mass vs. time: indicates directly core degradation progress	kg
DTsat in core	K
DTsat in Upper Head	K
Intermediate core flux	
H2 mass generated in core	kg

Table VII: Physical data regarding primary circuit behavior

Parameter	**Unit**
Primary circuit pressure	MPa
RPV pressure	MPa
Pressurizer Safety/Relief valve operation/flow rate	kg
High pressure SI mass flow rate,	kg
Emergency core injection flow rate	kg
Emergency water storage tank level	m
ECCS flow rate (HPCI/LPCI according to plant design and need)	kg/s
Low pressure SI mass flow rate,	kg/s
RHR flow rate	kg/s
Cold leg temperature	K
Hot leg temperature	K
Inlet header water level	m
RPV temperature	K
Water level in pressurizer	%
Charging/let down line mass flow rate	kg/s
Water level in tank for safety injection	m
FP retention in RCS/HTS	kg
Thermal exchange with safety systems	W
Mass flow rate at RCS break (break size)	kg/s
Feedwater flow rate	kg/s
Feedwater temperature	K
RCIC/AFW flow rate	kg/s
ADS flow rate	Kg/s
Suppression pool temperature	K
Suppression pool water level	m
MSIV status	
Steam flow rate in MSL	kg/s
SLCS flow rate	kg/s
Boron injection flow rate	kg/s
IC flow rate	kg/s

Table VIII: Physical data regarding secondary circuit behavior

Parameter	Unit
Secondary pressure	MPa
SG isolation (water and vapour),	-
Atmospheric valve mass flow rate	kg/s
Water level in SG	%
Feedwater mass flow rate to SG	kg/s
FP activity in SG	Bq
Number of break tube in case of SGTR – break size	-
Mass flow rate between primary and secondary in case of SGTR	Kg/s

Table IX: Physical data regarding containment behavior

Parameter	Unit
Reactor building pressure	MPa
Vacuum building pressure for CANDU	MPa
PCV pressure	MPa
Reactor building temperature	K
Vacuum building temperature for CANDU	K
Containment oxygen level	%
Containment hydrogen level	%
Containment activity	Bq
Total Flow rate of containment spray system	kg/s
Containment spray tank water level	m
Total Containment depressurization system flow rate	kg/s
Lower drywell water level	m
Water level in containment sump	m3
Water temperature in containment sump	K
Containment dose rate	Gy/s or Sv/s
Average wall temperature in containment	K
Average temperature in annulus space containment : for PWR1300	K
Annulus containment pressure : for PWR1300	MPa
Containment/drywell pressure	MPa
Containment/drywell temperature	MPa
Filtered containment venting system flow rate	kg/s
In-vessel hydrogen generation rate	Kg/s
Ex-vessel hydrogen generation rate	kg/s
Cavity temperature	K
Suppression pool temperature	K
Supression pool water level	m

Lower drywell water level	m
Mass of molten material in reactor pressure vessel pedestal and drywell	kg
Temperature of molten material outside reactor pressure vessel	K
Carbon monoxide generation by MCCI	kg

Table X: Physical data characterizing the release

Parameter	Unit
Mass flow rate in chimney	m3/s
Volumetric activity in chimney.	Bq/m3
Offsite radiation measurements	Sv/s
Leak rate of containment building to each building,	kg/s
Leak rate of auxiliaries buildings,	kg/s
Direct release from reactor building	Bq
Direct release from auxiliary building	Bq
Direct release from ventilation systems	Bq
Volumic activity in chimney	Bq/m3
FP release for each family and isotope from vessel to RCS,	kg
FP release for each family and isotope at the RCS break,	kg
FP concentration evolution in reactor building for each FP family and isotope	kg
FP deposited on wall (family and isotope),	kg
FP transfer in auxiliary building (family and isotope),	kg
FP mass on filter (family and isotope).	kg
Leak rate or flow rate of the containment depressurization system,	kg/s
Leak rate or flow rate by the raft after breakthrough.	kg/s
FP retention in RCPB	kg
FP mass retained in suppression pool	kg
Leak rate from containment to each building	kg/s
Leak rate of containment building to annulus space	kg/s
Leak rate of containment building to fuel building	kg/s
Leak rate of containment building to nuclear auxiliary building	kg/s
Leak rate of containment building to safety auxiliary building	kg/s
Leak rate of safety auxiliary building to environment	kg/s
Leak rate of nuclear auxiliary building to environment	kg/s
Leak rate of fuel building to environment	kg/s
Leak rate of containment building to environment	kg/s
Leak rate of containment building to environment (filtered)	kg/s
Direct release from safety auxiliary building	Bq
Direct release from nuclear auxiliary building	Bq
Direct release from fuel building	Bq
Direct release from reactor building	Bq
Direct release from reactor building (filtered)	Bq

Table XI: Limited element and isotopes to calculate release

NG release (% Bq)	Te-127m
I released (Bq or kg)	Te-129
Cs release (Bq or kg)	Te-129m
Xe-133	Te-131
Xe-135	Te-131m
Kr-85	Te-132
Kr-85m	Sr-88
I-131	Sr-89
I-132	Sr-90
I-133	Rb-85
I-135	Rb-87
Cs-134	Mo-95
Cs-136	Mo-97
Cs-137	Mo-98
Te-127	Mo-100

Table XII: Extended element and isotopes to calculate release

As-78	Cd-117m	I-130	Nd-147	Rh-103m	Sn-121	Te-129m	Y-91m
Ba-137m	Ce-141	I-131	Np-238	Rh-105	Sn-123	Te-131	Y-92
Ba-139	Ce-143	I-132	Np-239	Rh-106	Sn-125	Te-131m	Y-93
Ba-140	Ce-144	I-133	Pd-109	Rh-106m	Sn-127	Te-132	Zr-95
Ba-141	Cm-242	I-134	Pm-147	Rh-107	Sn-128	Te-133	Zr-97
Ba-142	Cm-244	I-135	Pm-148	Ru-103	Sr-89	Te-133m	
Br-82	Cs-134	In-115m	Pm-148m	Ru-105	Sr-90	Te-134	
Br-82	Cs-136	In-116m	Pm-149	Ru-106	Sr-91	U-237	
Br-82	Cs-137	In-117	Pm-150	Sb-124	Sr-92	U-238	
Br-82	Cs-138	Kr-85m	Pm-151	Sb-125	Tb-160	U-239	
Br-83	Cs-139	Kr-87	Pr-143	Sb-126	Tb-161	Xe-131m	
Br-83	Cs-140	Kr-88	Pr-144	Sb-127	Tc-101	Xe-133	
Br-83	Eu-154	La-140	Pr-145	Sb-128	Tc-104	Xe-133m	
Br-83	Eu-156	La-141	Pu-238	Sb-128m	Tc-99m	Xe-135	
Br-84	Eu-157	La-142	Pu-239	Sb-129	Te-125m	Xe-135m	
Br-84	Gd-159	Mo-101	Pu-240	Sb-130	Te-127	Xe-138	
Br-84	Ge-77	Mo-99	Pu-241	Sb-131	Te-127m	Y-90	
Br-84	H-3	Nb-95	Rb-86	Se-83	Te-129	Y-91	

Table XIII: Physical data characterizing iodine chemistry

Parameter	Unit
pH in sumps,	
Silver mass in sump,	kg
Sump dose rate,	Gy/s
Molecular iodine mass on paint,	kg
Organic iodine mass coming from adsorbed I2 on paint,	kg
I- mass in sump, suppression pool	kg
I2 released from sump, suppression pool	kg
Suppression pool activity	Bq

Table XIV: Physical data characterizing CANDU

Parameter	Unit
Moderator level	m
Moderator temperature	K
Cover gas pressure	MPa
Vacuum Building pressure	MPa
Calandria vault/Shield tank level	m
Calandria vault/Shield tank water temperature	K
Core debris mass in Calandria	kg

Table XV: Physical data characterizing SFP

Parameter	Unit
Water level in SFP	m
Water temperature in fuel pool	K
Heat exchange in spent fuel pool	W

3. USE OF THE DATABASE TO TEST FASTNET TOOLS

An exercise has been done to test the capability of FASTNET participants to use PERSAN and RASTEP to predict the source term for different scenarios. The predicted source terms are being compared against the source term database for these scenarios.

Four of them were selected for the exercise, that represent different reactor designs:

- VVER 440: a severe SBO accident scenario for a general VVER-440 reactor;
- ABB-II BWR: steam line break with loss of ECCS;
- CANDU: Single Unit SBO in a single unit containment with loss of ECCS;
- PWR 1300: LOCA, 6P SI and CSS lost at recirculation.

4. SEVERE ACCIDENT CODE APPLICATION AND UNCERTAINTY

Considering the complexity and mutual different interacting and interrelated phenomena/processes along a SA transient progression [15], the most accurate simulation tools to estimate the related source terms are the State-of-the-Art SA integral codes (e.g. ASTEC, MAAP, MAAP-CANDU, MELCOR, etc.). These codes having integrated all the knowledge developed in the last decades from the experimental activities, permit the prediction of the transient progressions of the plant considered. This allows characterizing the main SA phenomena taking place in the RPV, the reactor cavity, the containment, and the confinement buildings typical of NPP. Several models/correlations have been implemented in these State-of-Art SA codes and have to be set by code-user during input-deck development.

In relation to the code model/correlations implemented in the State-of-the-Art severe accident codes, even though several experimental campaigns in the field of SA phenomena [16-20] have been performed and provided valuable “assessment database [21]” as references to assess SA simulation tools, the analyses of the current State-of-the-Art shows that there is need to reduce some uncertainties still present [22]. However, consequent investigation of phenomena/processes, using geometric prototypical experimental facility with prototypical material, should be addressed. As a consequence, discrepancies in some of the core degradation phenomena can be still observed when comparing the results as predicted by different simulation tools considering the different core degradation models implemented in the codes [23]. Considering the need to reduce and/or evaluate some uncertainties still present and considering the reached level of development and maturity of SA codes and their application in the assessment of SAMG the discussion and application of SA progression analyses with uncertainty estimation is currently a key topic in the Best Estimate Plus Uncertainty (BEPU) framework [24].

To address the gap in the current experimental database and in modelling and simulations [25], many multinational efforts were launched to guide research and development efforts. Several research activities in national and international framework (European Commission framework [e.g. 26, 27], OECD/NEA/CSNI framework [e.g. 28, 29], IAEA framework [e.g. 30]) were performed or are in progress in order to reduce the epistemic uncertainty in SA phenomena and to prevent or mitigate SAs. In addition, comprehensive, in-depth code-to-code comparison exercises, also called “crosswalk” activities, led by the code developer teams have been performed in order to identify key modelling differences between the codes. Good examples are the MAAP-MELCOR [31] and the ASTEC-MELCOR [32] crosswalks.

In relation to the user-effect [33,34], considering the several complex and different phenomena/processes taking place during a SA, code-users require a high level understanding of the phenomena/processes and need to define several modelling parameters when using a SA code. It is also important to note that, in general, along the input-deck development the code-user need to find a compromise between the computational time and the degree of detail of the results and approximations are thus unavoidable.

Considering the above mentioned arguments, though the SA code give the State-of-the-Art answer to predict source terms releases, it is necessary to be aware that the results are affected by uncertainties related to models/correlations and user-effect. Considering the key role of SA code for deterministic safety analyses and source term evaluations, several research activities in national and international frameworks are in progress and are planned to reduce and/or estimate the uncertainty in SA phenomena prediction.

5. CONCLUSIONS

In the framework of the FASTNET project funded by the Horizon 2020 Framework Programme and coordinated by IRSN, the development of a common database of pre-calculated SA scenarios, trying to cover all concepts of existing plants in Europe and other participating countries (e.g. Canada) is one of the key activities of the project. Two main goals have to be addressed by this database. The first goal is related to test the methods for emergency response, and the second is to support the further development of fast running tools. The fast running tools identified at European Level to be part of the FASTNET methodology are the deterministic code PERSAN, developed by IRSN (France), and the probabilistic code RASTEP, based on Bayesian Belief Networks and pre-calculated source terms, developed by Lloyd's Register (Sweden). SA codes have been used to develop the database scenarios and therefore to estimate the transient scenario evolution and the potential related source term releases.

Along the paper the criteria adopted to pick the FOMs, to summarize the SA-code applications, and the applicability of this database for supporting fast nuclear emergency tools development have been discussed. A scenarios matrix covered by the database is presented. Each scenario has been described in a unified way to be included in the database and a special spreadsheet format has been used to collect the results. The list of the key parameters included in the spreadsheet, for the description of the accident scenario, has been presented in detail in the paper. During the project an exercise was conducted to test the capability of FASTNET participants to use PERSAN and RASTEP to predict the source term for four different scenarios using four different generic plant models. A brief description of scenarios used for the exercise is also presented in the paper.

It is to underline that at the end of the database development, the data base might be suitable for application by emergency management organizations in support of the estimation of the consequences to the public and the population protection measures. In this respect, an interface was developed using the IAEA IRIX [35] format to transfer the source term from the fast source term tools to the decision support systems. This interaction is tested and evaluated in the frame of the project. This provides for the first time a complete chain of interaction from the accident initiation through the accident analysis, accident progression, source term estimation and finally consequence assessment. The products developed in FASTNET can be further used by all end users running the tools and appropriate interfaces. Finally, it is foreseen to transfer the database, or its content, to IAEA, for long-term maintenance and further extension in terms of scenarios availability, for any future uses beyond FASTNET.

Though the SA codes, used to develop FASTNET database, provide State-of-the-Art predictions of a potential source term release, it is necessary to be aware that the results are affected by uncertainties related to models/correlations and user-effect. Therefore, in the final section of the paper has been in brief described the general source of uncertainties in SA codes. Considering the key role of SA codes for deterministic safety analyses and source term evaluation, several research activities in national and international frameworks are in progress and are planned to reduce and/or estimate the uncertainty in SA phenomena prediction.

NOMENCLATURE

ADS	Automatic Depressurisation System
ASTEC	Accident Source Term Evaluation Code
BDBA	Beyond Design Basis Accident
BEPU	Best Estimate Plus Uncertainty
CSS	Containment Spray Systems
ECCS	Emergency Core Cooling System
EP&R	Emergency Preparedness & Response
FCVS	Filtered Containment Venting Systems
FOM	Figure Of Merit
FP	Fission Product
HPSI	Loss of Feedwater in Steam Generator
IC	Isolation Condenser
IRSN	Institut de radioprotection et de surete Nucleaire (Institut for radiological Protection and Nuclear Safety)
LOCA	LOss of Coolant Accident
LEI	Lithuanian Energy Institute
LPSI	Low Pressure Safety injection
MAAP	Modular Accident Analysis Program
MELCOR	Methods of Estimation of Leakages and Consequences of Releases
NPP	Nuclear Power Plant
PORV	Power-Operated Relief Valve
PRISE	Primary-to-secondary leak event
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RWST	Refueling Water System Tank
SCRAM	Safety Control Rod Axe Man
SA	Severe Accident
SAMG	Severe Accident Management Guidelines
SBLOCA	Small Break LOCA
SBO	Station Black-Out
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SLCS	Standby Liquid Control System

ACKNOWLEDGMENTS

Authors thank the funding received from the H2020 Framework Programme of the European Commission via the FASTNET project; Special thanks to IRSN for the coordination of Project.

REFERENCE

[1] Severe Accident Management Programmes for Nuclear Power Plants, IAEA Safety Standards, Safety Guide No. NS-G-2.15.

[2] Bentaïb, H. Bonneville, G. Cénérino, B. Clément, F. Corenwinder, M. Cranga, G. Ducros, F. Fichot, D. Jacquemain, C. Journeau, V. Koundy, D. Leteinturier, D. Magallon, R. Meignen, F. Monroig, G. Nahas, F. Pichereau, E. Raimond, J-M. Seiler, B. Tourniaire, J-P. Van-Dorsselaere; D. Jacquemain,

Coordinator; Nuclear Power Reactor Core Melt Accidents, Current State of Knowledge, IRSN, EDP Sciences, France (2015).

[3] M. Pescarini, F. Mascari, D. Mostacci, F. De Rosa, C. Lombardo, F. Giannetti, "Analysis of Unmitigated Large Break Loss of Coolant Accidents Using MELCOR Code", *35th UIT Heat Transfer Conference (UIT2017), IOP Conf. Series: Journal of Physics*: Conf. Series 923 (2017) 012009.

[4] J. Fontanet, J. Panalva, L. E. Herranz, "Towards Fast Tools for Diagnosis/Prognosis in Severe Accidents. CIEMAT Contribution to the Scenarios Database within the FASTNET Project", *Sociedead Nuclear Espanola*, *44° Reunion Annual*, 26-28 September 2018.

[5] https://www.fastnet-h2020.eu/

[6] P. Chatelard, S. Belon, L. Bosland, L. Carénini, O. Coindreau, F. Cousin, C. Marchetto, H. Nowack, L. Piar, L. Chailan, "Main modelling features of ASTEC V2.1 major version", *Annals of Nuclear Energy*, Volume 93, July 2016, Pages 83-93.

[7] Fauske & Associates, LLC "Transmittal Document for MAAP5 Code Revision MAAP 5.02", FAI/13-0801, 2013.

[8] Fauske & Associates, LLC "Transmittal Document for MAAP-CANDU5.00A", FAI/18-0111, 2018.

[9] MELCOR Computer Code Manuals, Vol.1: Primer and Users' Guide, SAND 2015-6691 R; Vol. 2: Reference Manual, SAND 2015-6692 R; Vol. 3: MELCOR Assessment Problems, SAND 2015-6693 R; Sandia National Laboratories, USA (2015).

[10] G. Ortega-Nicaise, B. Quaghebeur, PERSAN user manual, FASTNET-RESP-P1, PSE-SANTE/SESUC/2018-00276, November 218.

[11] F. Di Dedda, Definition of a dynamic source term module within RASTEP, Chalmers University of Technology, 2013 REF2.

[12] F. Cousin, Relevant code data to define accident scenario progress, FASTNET.DATA.Dl.2, 2016.

[13] F. Cousin, J. Kubicek, F. Di Dedda, A. Rizoiu, Relevant code data to define accident scenario progress for emergency response and code development and validation, FASTNET.DATA.Dl.2 [Rev1], 2017.

[14] F. Cousin, K. Chevalter-Jabet, A. Ribermarklund, Reviewed specifications and methodologies for accident database, FASTNET.DATA.Dl.l, 2018.

[15] J.P. Van Dorsselaere, M. Barrachin, S. Power, M. Adorni, M. Hrehor, F. Mascari, A. Schaffrath, I. Tiselj, E. Uspuras, Y. Yamamoto, D. Gumenyuk, N. Fedotova, O. Cronvall, P. Liska, "The Summary of ETSON strategic orientations on research activities, ETSON Research Group activity", *ATW-International Journal for Nuclear Power*, Vol. 63 (2018), issue 1 January .

[16] Source term uncertainty, recent development in understanding fission product behaviour, NEA/CSNI/R(92)2.

[17] SOAR on Containment Thermalhydraulics and Hydrogen Distribution, NEA/CSNI/R(1999)16.

[18] In-Vessel Core Degradation Code Validation Matrix, Update 1996-1999, NEA/CSNI/R(2000)21, France (2001).

[19] State-Of-The-Art Report On Nuclear Aerosols, NEA/CSNI/R(2009)5.

[20] Containment code validation matrix, NEA/CSNI/R(2014)3.

[21] F. Mascari, H. Nakamura, K. Umminger, F. De Rosa, F. D'Auria, "Scaling issues for the experimental characterization of reactor coolant system in integral test facilities and role of system code as extrapolation tool, *Proceeding of International Topical Meeting on Nuclear Reactor Thermal Hydraulics 2015", NURETH 2015*, Chicago; United States; 30 August 2015 through 4 September 2015.

[22] J. P. Van Dorsselaere, J. Mustoe, S. Power, M. Adorni, A. Schaffrath, A. Nieminen, "ETSON views on R&D priorities for implementation of the 2014 Euratom Directive on safety of nuclear installations", Kerntechnik: Vol. 81, No. 5, pp. 527-534.

[23] L. Humphries, MELCOR Code Development Status EMUG 2018, *10th Meeting of the European MELCOR User Group*, University of Zagreb, Zagreb, Croatia, Faculty of Electrical Engineering and Computing – FER, April 25-27, 2018 https://www.psi.ch/emug/Emug2018EN/EMUG_2018_01.pdf.
[24] M. D'Onorio, F. Giannetti, F. Mascari, G. Caruso, "Uncertainty Analyses Using The RAVEN Software Tool Coupled With MELCOR Severe Accident Code", *Proceedings of ANS Best Estimate Plus Uncertainty International Conference (BEPU 2018)* Real Collegio, Lucca, Italy, May 13-19, 2018, BEPU2018-282
[25] J. C. de la Rosa Blul, S. Brumm, F. Mascari, S. J. Lee, and L. Carenini, "ASTEC–MAAP Comparison of a 2 Inch Cold Leg LOCA until RPV Failure", *Science and Technology of Nuclear Installations Volume 2018*, Article ID 9189010, 24 pages https://doi.org/10.1155/2018/9189010
[26] H. Nowack, P. Chatelard, L. Chailan, S. Hermsmeyer, V. Sanchez, and L. Herranz, "CESAM – Code for European Severe Accident Management, EURATOM project on ASTEC improvement", ERMSAR, Warsaw, Poland, 2017.
[27] HORIZON2020, "In-Vessel Melt Retention Severe Accident Management Strategy for Existing and Future NPPs," https:// cordis.europa.eu/project/rcn/196923 es.html.
[28] Organization for the Economic Co-Operation and Development (OECD), Nuclear Energy Agency (NEA), Committee of Nuclear Safety Installations (CSNI), "Informing Severe Accident Management Guidance and Actions through Analytical Simulation," Report on the WGAMA Working Group, 2017(ongoing as of June 2018).
[29] Organization for the Economic Co-Operation and Development (OECD) on State-of-the-Art Report on Molten Corium Concrete Interaction and Ex-Vessel Molten Core Coolability, NEA/CSNI/R(2016)15.
[30] International Atomic Energy Agency (IAEA) Technical Meeting on the Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors, October 2017, https://www.iaea.org/NuclearPower/Meetings/2017/2017-10- 09-10-12-NPTDS.html.
[31] Electric Power Research Institute, "Modular Accident Analysis Program (MAAP)MELCOR Crosswalk, Phase 1 – Study," EPRI Report, 2014.
[32] S. Belon, C. Bouillet, H. Bonneville,N.Andrews, and C. Faucett, "Insight of Core Degradation Simulation in Integral Codes Throughout ASTEC/MELCOR Crosswalk Comparisons and ASTEC Sensitivity Studies," *Proceedings of the ERMSAR 2017*, Warsaw, Poland, 2017.
[33] NEA/CSNI/R(98)22: Good Practices for User Effect Reduction, Status Report, November 1998-
[34] Approaches and Tools for Severe Accident Analysis for Nuclear Power Plants, IAEA Safety Reports Series No. 56, IAEA, Vienna, 2008.
[35] International Radiological Information Exchange (IRIX) Format, Version 1.0 Reference Description, IAEA Incident and Emergency Centre, 2013-03-31.