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REDUCTION OF RADIOLOGICAL ACCIDENT CONSEQUENCES

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Abbreviations

BWR	Boiling Water Reactor
CVCS	Chemical and Volumetric Control System
DBA	Design Basis Accident
DEC-A	Design Extension Condition type A
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
EPR	European Pressurized Reactor
FP	Fission Product
HBU	High Burnup
LOCA	Loss Of Coolant Accident
LOOP	Loss of Offsite Power
NPP	Nuclear Power Plant
PWR	Pressurized Water Reactor
RC	Radiological Consequences
REX	Return of Experience
SF	Single Failure
SI	Safety Injection
SGRV	Steam Generator Relief Valve
SGTR	Steam Generator Tube Rupture
SLB	Steam Line Break
SCS	Shutdown Cooling System
TH	Thermalhydraulic
VVER	Water-Water Energie Reactor *(Vodo-Vodjanoi Energetitsjeski Reactor)*

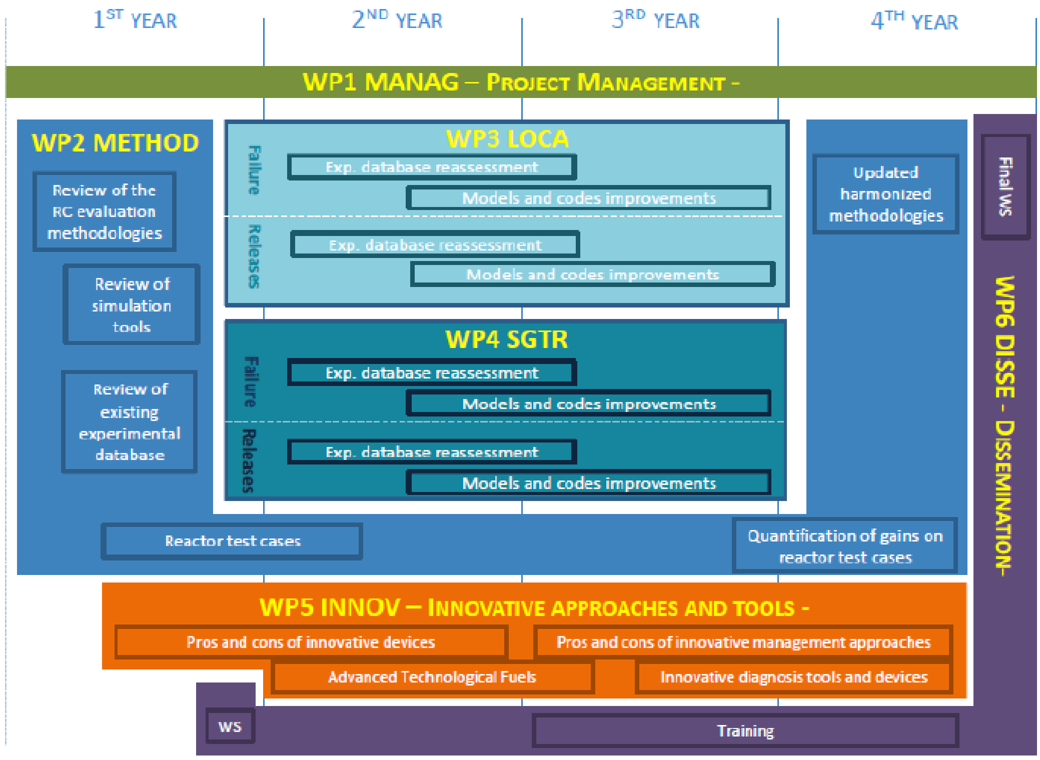
1 Introduction

This document is situated within work-package 2 (methodologies) of the R2CA project. The main goals of the R2CA project are:

- ✓ To reduce the degree of conservatism in safety evaluations of LOCA and SGTR bounding scenarios within DBA and DEC-A conditions by improving the simulation tools and the calculation methodology,
- ✓ To increase the safety level by optimization of the accident management actions and by development of innovative devices and tools for a better accident management and their anticipated diagnosis.

The main goal of the work-package 2 is to propose harmonized methods for evaluation of the radiological consequences of both SGTR and LOCA categories of DBA and DEC-A accidents. And the present deliverable 2.8 is dedicated to this task (task number is 2.6 and deliverable is 2.8 due to a gap in the numbering along the project).

It was decided at the beginning of the project that each partner could chose his own methodology, the added value of the project being to identify and to quantify the methodologies improvements that are specific for each partner. Therefore, the qualifier "harmonized" as original conserved title of the present deliverable is not representing what has been obtained at the end but anyway it content plenty of ideas of evolutions for each partner.



The above figure is issued from the grant agreement [1] and shows the place of the present deliverable in the project.

In this figure, the blue zone defines the WP2 : at the beginning of the project, a review of the RC methodologies of the different partners was written as D2.1 [2].The present document appears as the evolution of this D2.1 at the light of the progress that were made by each partner.

The present subject is multidimensional because concerns:

- Different kinds of reactors: PWRs, VVERs, BWR, EPR
- Two kinds of accidents with their own specificities: SGTR and LOCA

- For a kind of accident, a severity of the initiator affecting the licensing rules that are related to the study: DBA or DEC

Therefore, as the present deliverable is based on many deliverables that were produced as input in the frame of the present project, a compromise has been necessary in order to produce a valuable synthesis work which can be exploited for the future.

2 Main features of the LOCA and SGTR accidents

The aim of this chapter is to briefly go through these two events and have a first rough and general view of both accidents and the ways they can lead to radiological consequences.

2.1 LOCA: description of the accident and its source term

LOCA is the most significant SAR §15 accident study and was considered as dimensioning for PWR's since their origin.
The plant is initially considered at nominal power.
The initiating event is a break somewhere in the primary circuit (for PWRs: not in the SG, otherwise the event is defined as an SGTR). The location and the size of the break strongly influences the characteristics and the dynamic of this accident, presenting several complex thermal hydraulic phenomena and for which a lot of studies have been performed. This is not here the aim to make here such description.

The reactor trip occurs usually fast by primary pressure reduction and turbine trip occurs as a consequence.
ECCS start is not able to avoid the core uncovery for large break sizes. A loss of offsite power can be potentially considered at scram, leading to primary pump coast down, and need for diesel to feed safety equipment.

Regarding core cooling, the safe state is obtained when the primary system is at atmospheric pressure, the water that escaped from the primary system to the containment is collected in sump and reinjected after being cooled through SCS heat exchangers.

About the radiological consequences: as postulated by default or as a function of the fuel thermal boundary conditions during the event, specific damage will appear what will result in FP releases into the primary circuit. The different kind of damages to the fuel rods is represented by the below picture.

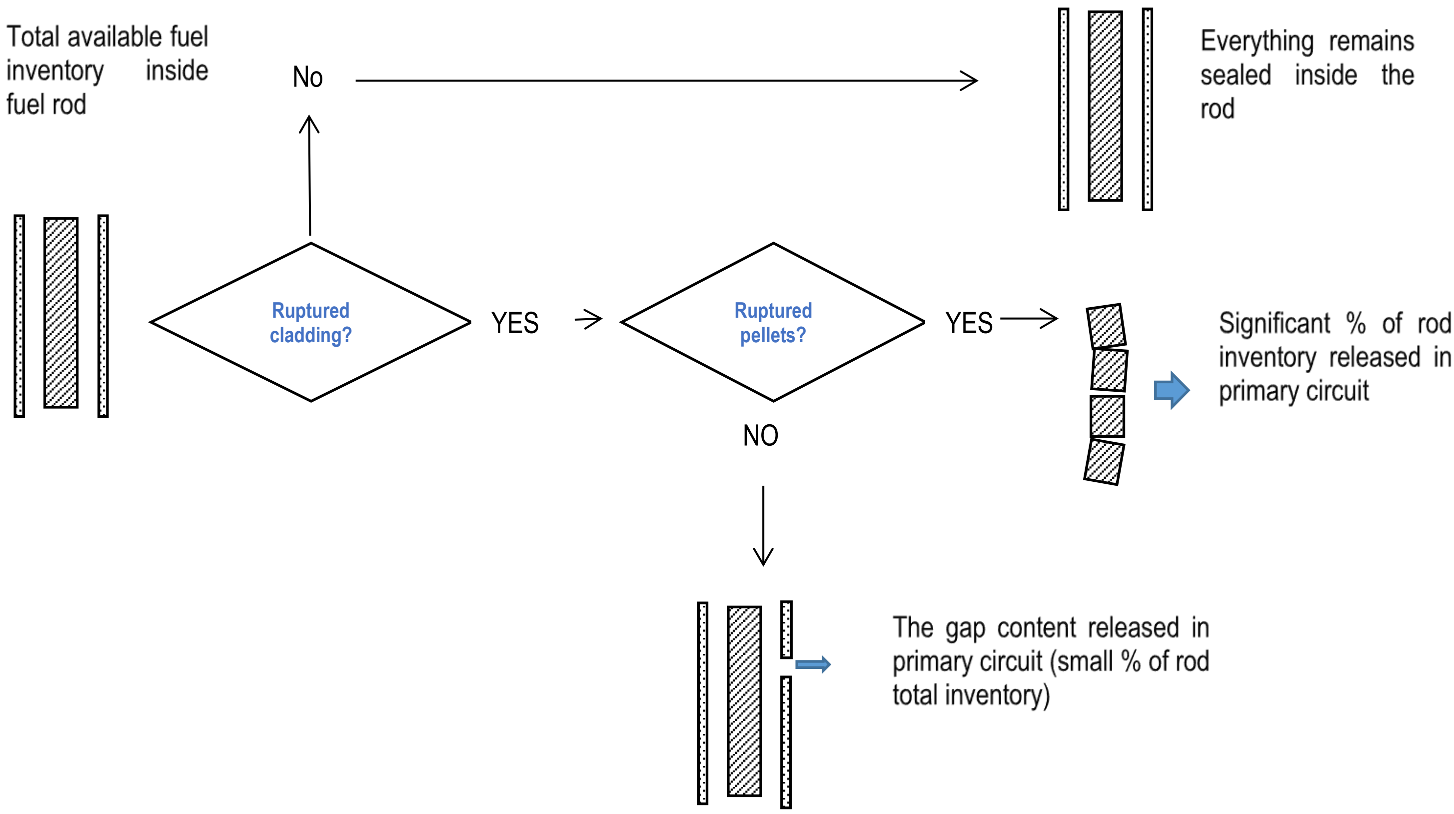


Figure 1 : Different potential status of fuel rods following a LOCA

As a function of this damage, different kinds and amounts of fission products (FP) will be released in the primary coolant. A strict and rough conservative evaluation of the source term imply damaged fuel pellets but in the frame of the current R2CA project, only clad burst is considered (i.e. the conditions leading to such clad bust and the resulting releases).

The contaminated primary water is injected in the containment through the break. Except for noble gases, and to a less extend gaseous iodine, thanks to the containment spray, the activity which is present in the gas phase (due to aerosols) inside the containment can be quickly reduced (accelerated settling).
Contamination goes to the environment through containment potential paths of leaks, filtered or not filtered, or through the stack (after collection) in case of specific procedures which is better for the atmospheric dilution.

2.2 SGTR: description of the accident and its source term

As a first important remark, note that such accident doesn't exist for BWRs.
Contrarily to LOCA, SGTR was not considered as an important event at the time of the Gen II PWR design (70'ies, 80'ies), but gained consideration with the REX and the number of fuel defective rod detected, such as significantly influencing the PWR's new designs like Gen III EPR.

The initiating event is the break of one or several SG tube(s), with various potential location that can have a significant influence on the phenomena that can take place. For VVERs, considering the design specificities of

their SGs compared to PWRs (horizontal vs vertical), another specific initiating event was considered which is the SG cover lift-up.
The reactor trip can happen by low primary pressure or Delta T signal, leading to start the diagnosis procedure by the operator. Safety injection will start in order to compensate the break flow, which is much smaller than for a LOCA.
At the secondary side, the turbine closes, and the steam dump is considered as not available what leads to a discharge through the atmospheric SG relief valve to evacuate the produced steam.
The SG main feedwater and auxiliary feedwater can be stopped automatically by high SG level setpoints, but the SG break flow can only be stopped if the pressures of the primary and secondary circuits are in equilibrium.
To do that, the operator has first to isolate the affected SG from the others, by closing the main steam isolation valve. Then, the operator will start the cooldown of the primary circuit by opening the intact SG relief valves. A failure to open a SG relief valve of one intact SG can be potentially considered as postulated single failure for the study.
After cutting the safety injection, the operator must depressurize the primary system by using either the pressurizer discharge valves, or pressurizer spray, up to the cancellation of the break flow. The study is usually finished (controlled state) when this break flow is cancelled because no more contamination can escape from the primary system, while in reality, there are still actions to perform before to go to a safe state with a primary system at atmospheric pressure.

The radiological analysis of such transient is very specific and can be complex as shown in the below picture, mainly characterized by the fact that the break is covered by water (flooded) or rather uncovered (the below picture refers to this second case).

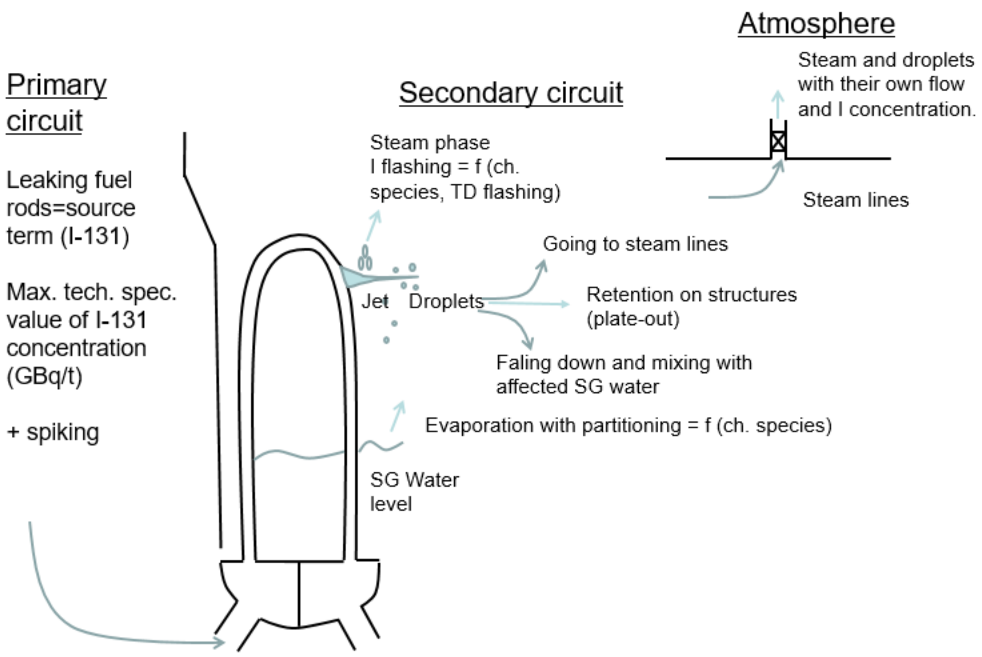


Figure 2 : Paths of releases during SGTR for an uncovered SG tube break

To model such kind of scenario, initially, a specific activity (Bq/m^3) is considered in the primary system for different species. This implies the presence of leaking fuel rods in operation, and/or also to tramp uranium (i.e. the presence of uranium traces elsewhere than inside fuel rods). A maximal value being imposed in the technical specifications of NPPs, a bounding SGTR study can consider such limit values as conservative initial state.
At transient initiation, a so-called “spiking” is supposed to start, during which the leaking fuel rods will release a part of their available radioactive inventory (contained in the gap between pellet and cladding) in the primary coolant, during a certain period. Other species than iodine can be also considered but knowing that as for iodine,

their behavior can be different than from the LOCA due to the fact that the cracks of the cladding are smaller than in the case of a “clad burst”.

Apart from this spiking resulting from already leaking fuel rods, usually no fuel damage is considered as a consequence of the event itself. This is a major difference compared to RC calculations of LOCA for which fuel damages is usually considered as a consequence of the event as shown in the previous chapter.

The spiking itself is both influenced by the primary pressure reduction due to the loss of coolant and by the scram. The predominance of the phenomena to be considered depends on the characteristics of the break (covered or uncovered) which in turn depends on:

- SG designs (vertical vs horizontal), hen on reactor models (PWRs vs VVERs). Indeed, for VVER’s most of the transients are related to covered (flooded) breaks.
- Break location for PWRs which can occurs near the top of SG tube bundle (uncovered/flooded break) or at tube bundle bottom (covered/flooded break).

Regarding this last point, for a PWR, a break near the top of the tube bundle can indeed enhance the formation of aerosols and then their transport to environment. Also in this case, the partitioning plays a role at the break location as part of the break flow is instantaneously transformed into steam (flashing phenomenon). This phase change especially affects the iodine repartition in both phases, depending on the iodine speciation in the primary circuit.

Leaks and smaller breaks are usually not simulated because can be compensated by the CVCS system and are managed by specific procedures.
In some cases, it remains possible that the CVCS system manages to compensate the break flow resulting from a single tube double ended break.

For covered breaks scenario, the overfilling of the affected SG must be avoided in order to preclude contaminated water release to the environment (most penalizing scenario in terms of environmental releases). Moreover, if such water discharge happens through SG relief valve or safety valves equipment that are not classified to discharge water, a break of such device has to be considered as a direct consequence of the initiating event, what leads to a massive, contaminated water release to the environment.
Without such overfilling, only contaminated steam from the affected SG can be released to the environment.

As a remark, the releases to environment during a SGTR are not occurring through the stack which is an inconvenient for dilution in the atmosphere. That’s also why SGTR is recognized as a potential “containment by-pass” event, because in most scenarios they lead to a by-pass of this barrier.

3 RC evaluation methodologies

As mentioned in the introduction, as the present deliverable 2.8 aims to be self-sustained and is a direct evolution of the deliverable 2.1, it was judged valuable to summarize here its content and to keep it in mind for the rest of the lecture.
The content of the present chapter gives a partial view of the document D2.1 [2] and summarizes the used methodologies for RC evaluations, at least for some of the partners. Different partners filled out a template that allowed to perform a comparison between them.

3.1 LOCA RC calculation methodology

3.1.1 Source term

The starting point is the well-known “source term”: the total inventory of radioactive material that can potentially be released to the environment, most of this inventory being inside the fuel rods of the core. The way to calculate this source term varies between partners: different codes (ORIGEN, SCALE, VESTA) can be used, or the isotopic inventory can be provided by the fuel supplier.

Figure 1 of present report illustrates the difference in noble gas releases from fuel to containment depending on organisations and on the hypotheses considered for fuel rod damages during the LOCA transient what constitutes the starting point of the propagation of the source term.

Using column chart in [2] allows to visualise which part of this original source term can propagate through different barriers up to the environment. The 100% starting point can be related to a given FP (i.e. Iodine), or to a group of FP with similar behaviour (i.e. noble gas).
At the end, it's clear that in most cases, only a very low amount of this source term is allowed to reach the environment, explaining the use of logarithmic scale for the column charts.

3.1.2 Propagation up to the containment

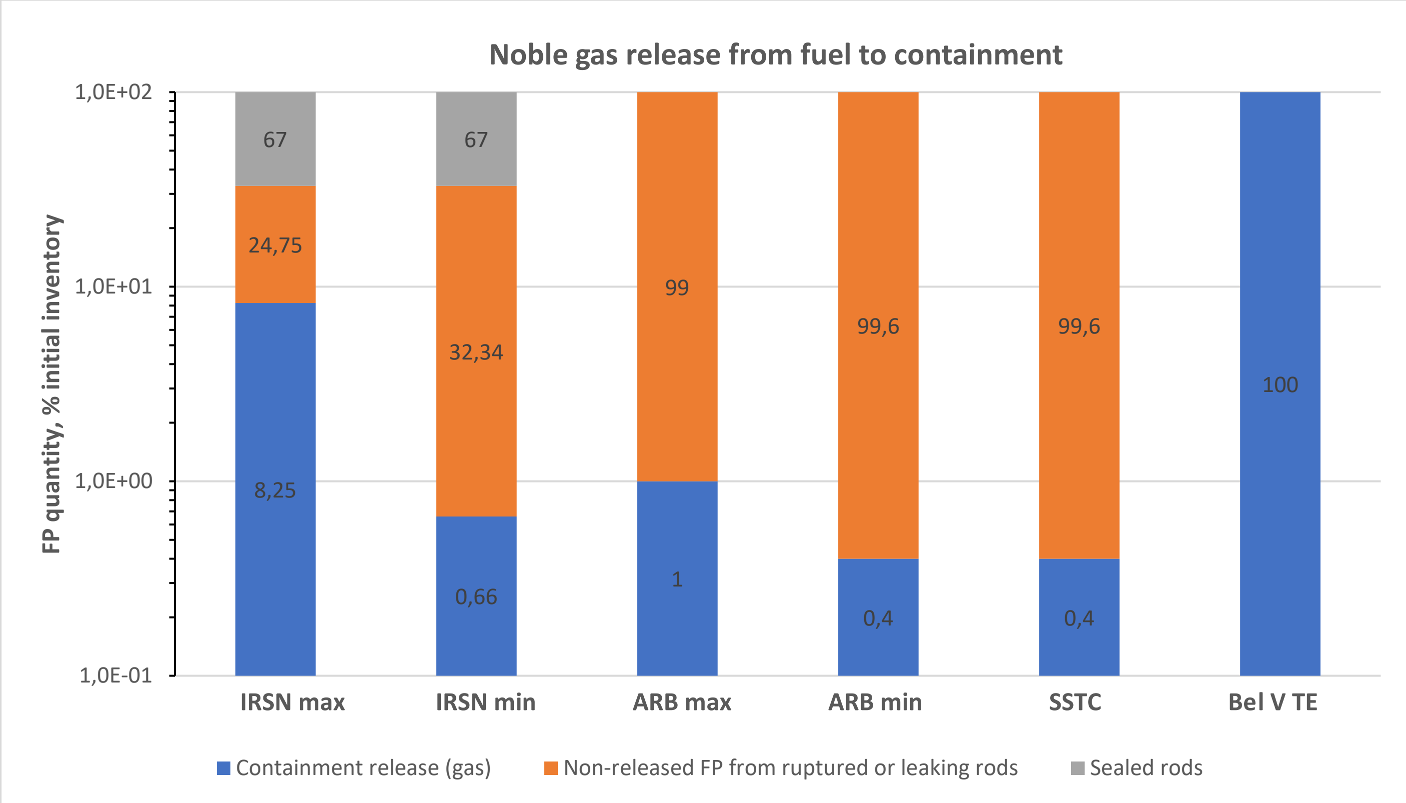
During the LOCA, the whole content of the primary coolant is supposed to be released to the containment and no retention (also called "plate-out") of contamination in the primary circuit is generally postulated by the partners.

A first important distinction is made between noble gases (Xe, Kr) and the other FP that can be more of less volatile.

A first column chart is obtained below (reproduced figure 1 of D2.1) which quantifies the noble gas (Xe, Kr) release to the containment (taking into account the failed fuel fraction and the FP release rate). Strong differences between partners already appear at this step.

This is explained in more details for some cases:

- For IRSN max (max and min are distinguished for conservative and less conservative evaluation), 1/3 of the fuel rods is considered as severely damaged and 25% of this inventory is released (while the rest stays inside the damaged fuel). This leads to 8.25% of the total inventory which is released to the containment, 24.75% which stays inside the damaged rods, and 2/3 (66%) of the inventory that remains sealed inside intact rods
- For ARB max, all the fuel rods are leaking through their cladding as a result of the LOCA, and 1% of their inventory in noble gas is considered to be released
- For Bel V/TE, 100% of the fuel is supposed severely damaged as a result of the LOCA and all of their inventory is considered as released (as if the whole core was transported in the containment), what represents the bounding case, this approach being possible for the licensing in Belgium due to the systematic use of a double containment.



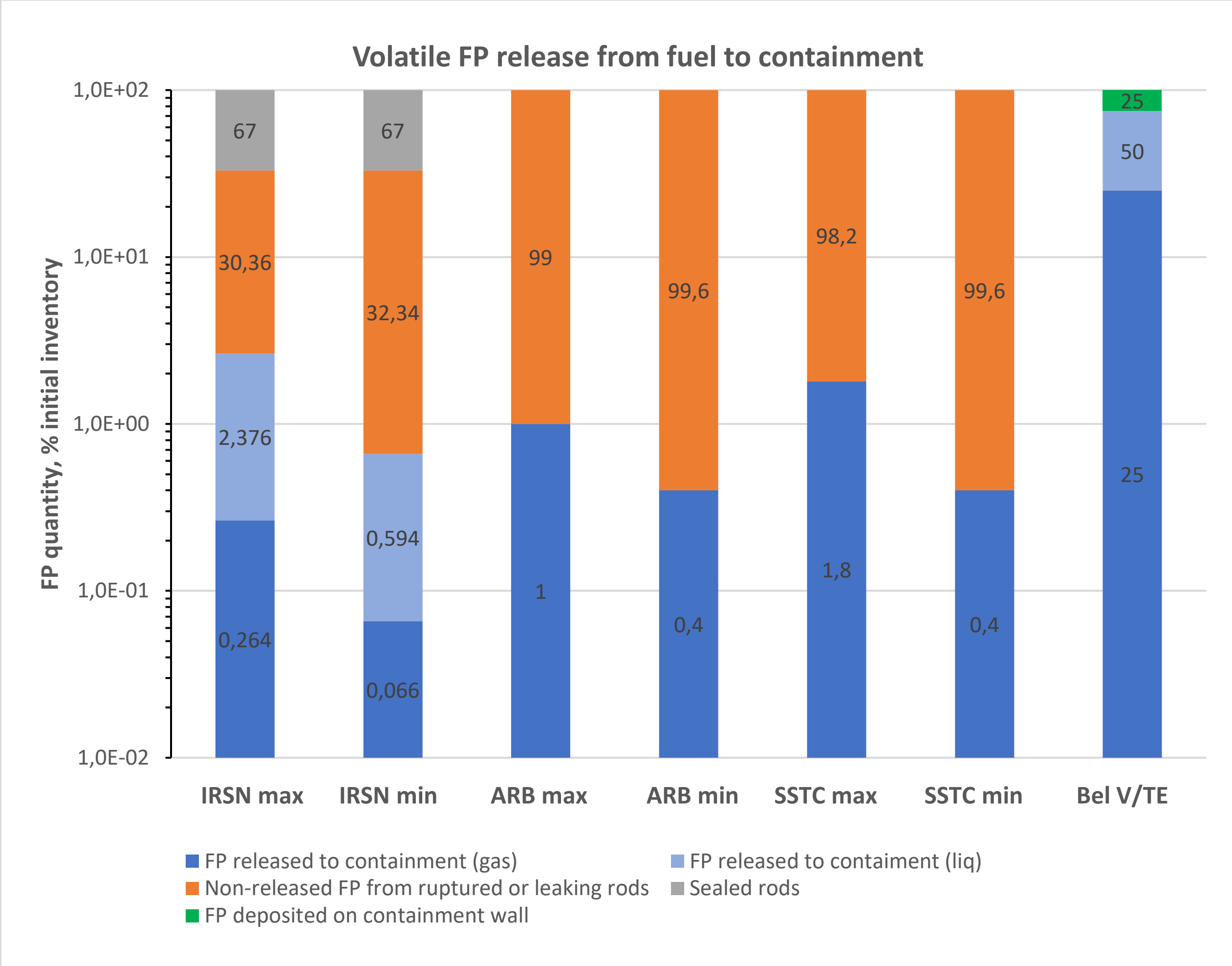
Reproduced figure 1 of D2.1

A second column chart is obtained below (reproduced figure 2 of D2.1) which quantifies the (more or less) volatile FP release to the containment and is a bit more complex than the one shown in Figure 1. Indeed, for the previous figure, 100% of the noble gas released goes to the gas phase of the containment. Here, the volatile FP which is released from the fuel is spread into 3 parts, implicitly taking into account the physical form of the volatile FP considered (vapour or aerosols) and their behaviour through their transport up to containment (i.e. depletion mechanisms due to condensation, diffusiophoresis, thermophoresis...) :

- A part that goes in the gas phase of the containment
- A part that goes in the liquid phase of the containment (IRSN being the only partner explicitly considering aerosols in the containment)
- A part which is captured by the surface of the containment (plate-out).

This is explained in more details for some cases:

- For IRSN max, 1/3 of the fuel rods is considered as severely damaged (this remains of course the same as for noble gas) and 8% of this inventory is released (while the rest stays inside the damaged fuel). Moreover, it is considered that 10% of this released part goes to the gas phase while 90% goes to the liquid phase. This explains that 10% of 8% of 33%, or 0.264 % of the source term goes to the gas phase
- For ARB max, as for noble gases, all the fuel rods are leaking as a result of the LOCA, and therefore 1% of their inventory in volatile FP is released. But ARB considers that 100% of this amount goes into the gas phase of the containment and therefore at the end, there is more volatile FP in the gas phase than for IRSN max calculation
- Bel V/TE is the only that takes credit of plate-out in the containment at this stage (25% of the whole source term).



Reproduced figure 2 of D2.1

3.2.2 Propagation from the containment up to the environment

The last part of the calculation consists in determining how the contamination can propagate from the containment up to the environment.
The containment represents a system in which several phenomena can take place: chemical reactions, settling of activity by the containment spray, plate-out, …all of these being time dependent. Different strategies are used by the partners, with significant variability in terms of complexity (a simple and bounding approach is often used when possible in the frame of licensing of NPP's).
Codes like SOPHAEROS are used by IRSN and LEI to model chemical reactions and aerosol behaviour/depletion.

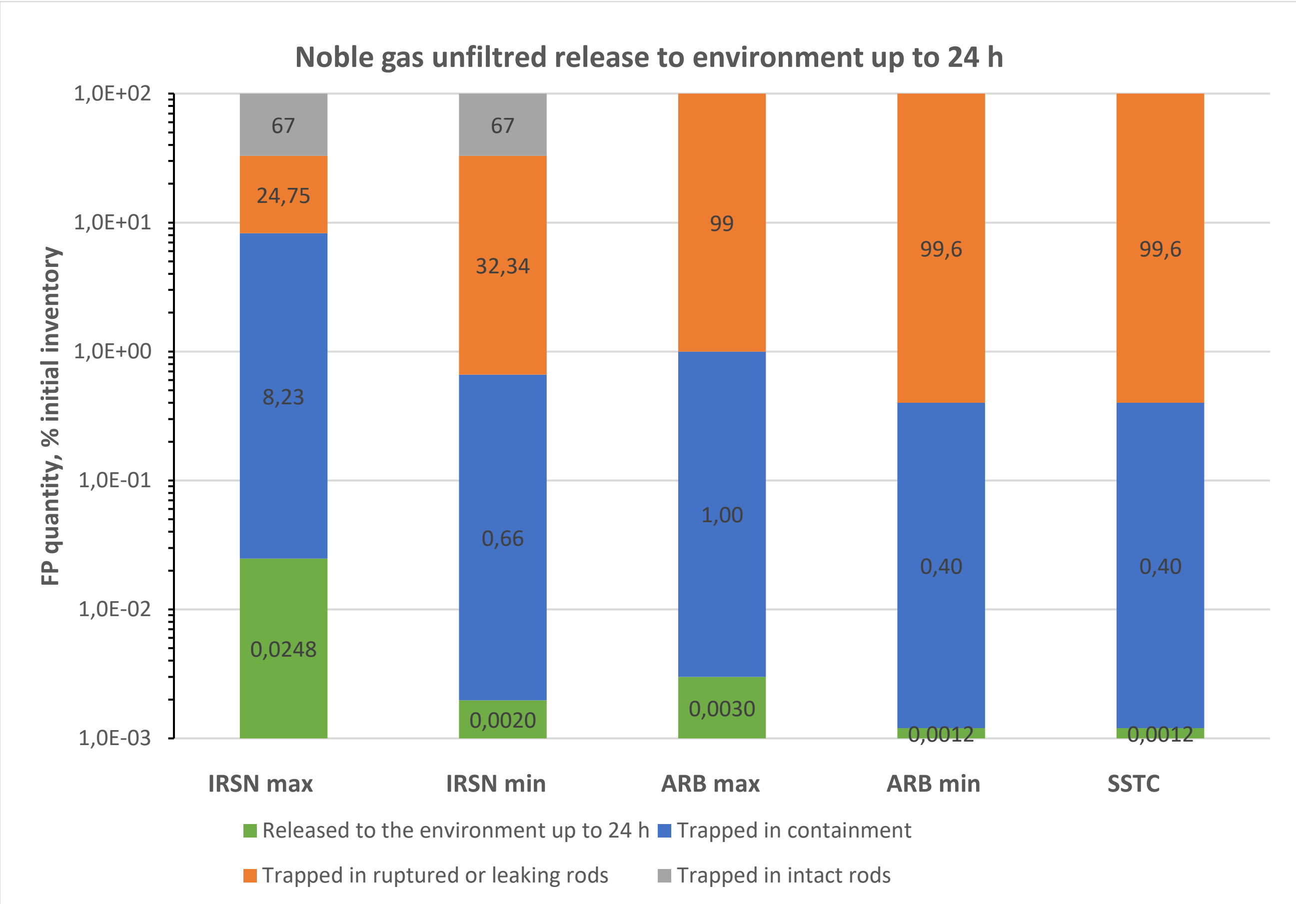
The containment spray acts on both phases of the volatile species (vapour and aerosols) that are present in the containment gas phase:

- The droplets of spray will reach an equilibrium with the vapour species originally present in gas phase, absorbing a part of their activity. The distribution between the containment gas phase and the droplet liquid phase will partly depend on their speciation and partition coefficient. Though two different injection flow rates for the spray over the whole transient are generally considered depending on the spray mode (injection of recirculation spray mode), an exponential law of contamination reduction is chosen: considered by IRSN, Bel V and TE.

- The contaminated aerosols will be settled: only modelled by IRSN with two different exponential laws of contamination reduction.

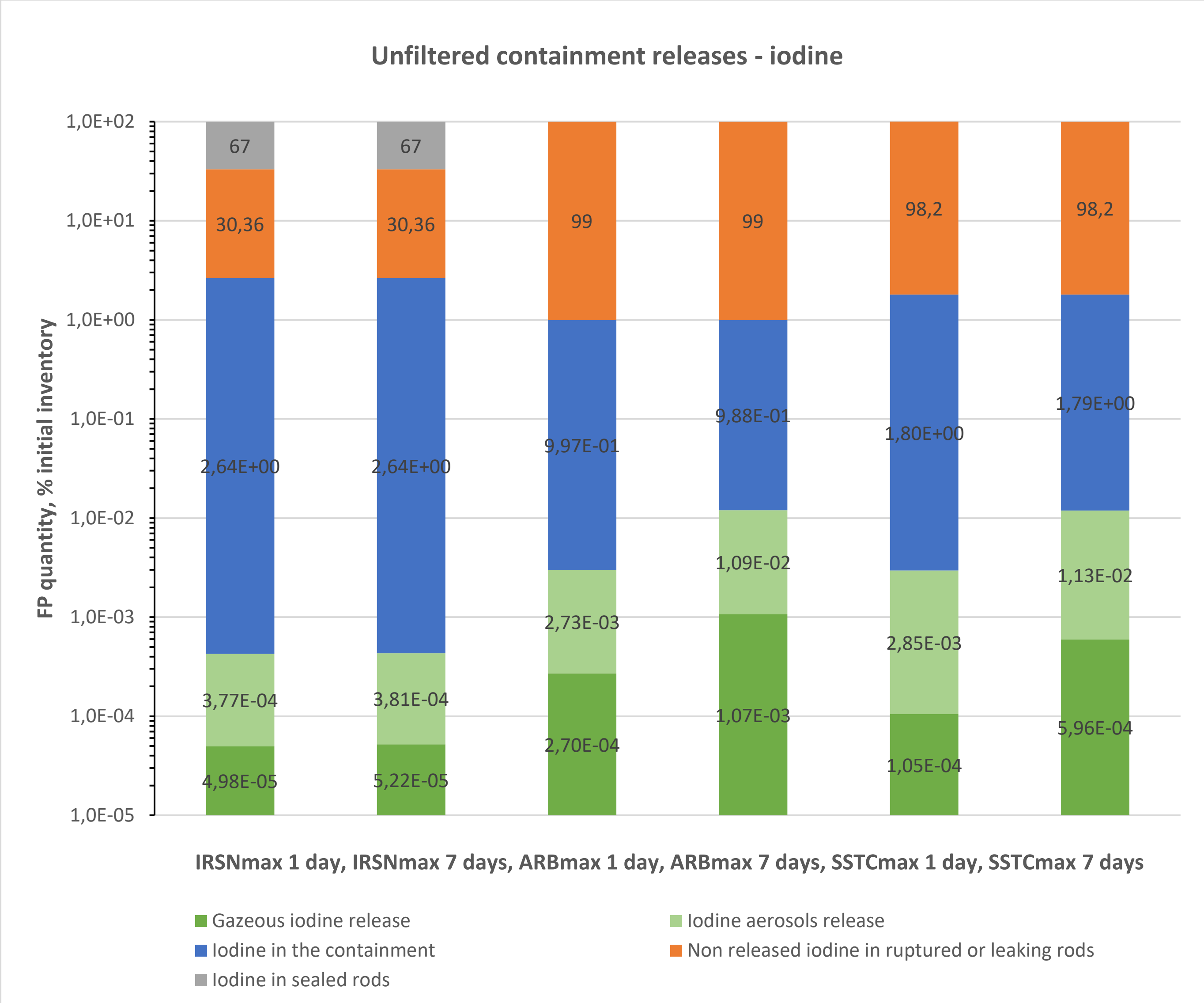
Unfiltered and filtered releases have to be considered.

The calculation is the most simplest for the noble gases because there is no mean to “reduce” such contamination (no plate-out, no settling, no capture by the spray, no filtering). Therefore, the amount which is released to the environment depends on the leak rate of the containment (and/or auxiliary building whenever considered.
The (reproduced from D2.1) figure 6 below can be directly obtained from figure 1 of D2.1, considering 0.3% leak per 24 h for single containment (Bel V/TE contribution is not added because of systematic use of double containment in Belgium, for which the leak through penetrations in both containment is very low).
In figure 6, unfiltered noble gas release after 1 day represents from 1.2E-3 % to 2.5E-2 % of the initial inventory.



Reproduced figure 6 of D2.1

Figure 8 represents unfiltered iodine release for a period of 1 day and 7 days.
Compared to figure 2, only a small part of the iodine inside containment will be released to environment (mostly in aerosol form). The calculation is more complex than for noble gas because the iodine concentration inside containment is time dependent.



Reproduced figure 8 of D2.1

The final figure 9 (not reproduced here) is obtained combining filtered and unfiltered iodine releases for a period of 1 day and 7 days. Filtered releases from the containment depend strongly on the decontamination factor of the filters (ranging from 1 for noble gases to 1000 for aerosols), and in general (for single containment units) this contributor is significantly smaller than unfiltered releases. This contributor was not considered in R2CA reactor calculations.

As a conclusion it can be observed from figure 8 that the fraction of the iodine source term that will be released to the environment goes from 5E-4 % to E-2 % depending on the partner and on duration of the release.
Just to give an order of magnitude (not written in report D2.1) : taking 85 E6 Ci (3145 E6 GBq) as typical value of iodine 131 source term in a PWR core, a fraction of 5E-4 % represents 15 700 GBq that are released to environment.
Taking typical 17x17 fuel assembly pattern inside a core containing 157 assemblies, this represents 264 x 157 = 41448 fuel rods, therefore:

- A fraction of 5E-4 % which is released to the environment represents 1/5 of a single fuel rod inventory
- A fraction of E-2 % which is released to the environment represents the inventory of 4 fuel rods

3.2 SGTR RC calculation methodology

3.2.1 Primary loop activity in steady state and during transient

The activity of the primary loop coolant is evaluated differently depending on the reactor concept and the partner. For VVER reactors, a wide range of FP is considered while for PWR's, different approaches were used: some of the partners only consider I-131 as being the only FP which is considered for the calculation while others consider not only iodine (different isotopes) but a larger list of FP including for the most important ones in terms of activity (i.e. several isotopes for nobles gases and caesium).
The conservative initial (steady state) situation is the operation of a NPP with a maximum iodine activity as authorized by the technical specifications (i.e. with the presence of leaking fuel rods) and penalized NPP operational feedbacks for the other isotopes whenever considered.
For the current project, partners used their own specific values of initial activities. Starting from these values, the SGTR transient will induce a so-called spiking of iodine, as introduced in previous § and as more detailed below (later in the project).

3.2.2 Iodine form in the primary loop

Characterizing the chemical form of iodine (speciation) in the primary system is interesting because this affects its behaviour in the transfer between barriers, and so the way it can be spread to environment.
This characterisation into different species is performed by IRSN, but not by other partners like TE and Bel V.
For VVER, the speciation is the same as for LOCA methodology.

3.2.3 Transfer in the steam generator

Like for the transfer to the containment for the LOCA, the transfer of FP to the gas phase of the SG during SGTR depends on its nature: noble gas, more or less volatile FPs, aerosols.
Contrarily to the containment for the LOCA, here the secondary part of the SG initially contains liquid water and steam into which the FP's (traveling through the tube break) will be transferred.
§4.3 of D2.1 [2] describes the different used models by the partners.
The way this transfer is performed depends among others on the localisation of the SG break compared to the level of water which is present in secondary side : a break which is situated near the top of the tube bundle is in contact with steam (and therefore is more susceptible to transfer FPs to gas phase through jet flashing and atomisation) while a break near the tube plate is most probably in contact with liquid water where specific phenomena such a pool scrubbing may play an important role (and therefore the transfer in that case will be mostly through partitioning and carry-over). These processes are also different for PWRs compared to VVERs due to their specific SG geometries and then, in turn, to the accidental scenarios considered.
As mentioned in §4.3 of D2.1 the balance of activity must be respected (everything going through the break must be found in both phases of the secondary side).

3.2.4 Secondary loop retention

From the secondary system, the easiest pathway to environment are the relief or safety valves where gaseous and/or liquid releases can occur in case the failed steam generator is overflowing or not.
The last figure 12 of D2.1 [2] (not reproduced here) represents the part of the activity transferred from the break into environment. The only retention which is considered by the partners is the retention in the liquid phase of the SG (i.e. droplet retention in the upper steam parts such as the swirl-lane or the chevron separators are not considered). In the report, a wide range of situations was observed as a function of the partner and the objective of the calculation (realistic or conservative).

4 LOCA and SGTR initial simulations: reactor test cases

Technical individual reports have been issued by each partner including performed simulations, hypotheses, and calculation results. These reports are part of the task T2.3 "Reactor test case simulations" and are summarized in deliverable D2.5 [9]
This first set of reports is using reference models and methodologies.

The table 1 below describes all performed simulations by reactor type and by type of transient (LOCA or SGTR).
A few remarks/observations:

- Several types of LWR reactor considered: PWR-900, PWR-1000, PWR-1300, PWR Konvoi, EPR, VVER-440, VVER-1000, BWR-4
- DBA and DEC scenarios can be the object of a single report, or of separated reports
- There was no calculation of SGTR for EPR

	PWR LOCA	PWR SGTR	EPR LOCA	EPR SGTR	VVER LOCA	VVER SGTR	BWR LOCA
ARB					[12] [4] [14] [15]	[16] [17] [18] [19]	
BEL V		[20][21]					
BOKU		[32] [33]				[34] [35]	
CIEMAT		[22][23]					
EK					[24]	[25]	
ENEA	[26][27]						
HZDR	[28]						
IRSN	[29][30]	[31]					
LEI							[36]
SSTC					[37][38]	[39][40]	
TRACTEBEL (TE)		[41][42]					
UJV					[43]	[44]	
VTT			[45]		[46]		

Table 1: Partners initial reactor test cases

4.1 Overview of the initial LOCA simulation reports

All chapters dedicated to LOCA are the §2.2.1, §2.3.1 §2.4 and §2.6.1 of D2.5 [9].
See also the §2.1 of the present deliverable.
The LOCA TH simulation usually requires a model of the primary and secondary systems and also a model of the containment.

The DBA and DEC A scenarios have been fixed by the task 2.2 of the project.

DBA simulations generally use conservative initial and boundary conditions, a single failure, and no operator action. A large break in the cold leg was often considered, even if some partners are analyzing in addition smaller breaks. A LOOP can be combined at some moment (scram signal). Initial states are at full nominal power.

In general, DEC-A scenarios were similar to DBA scenarios considering in addition one additional failure, except for two cases where different scenarios were considered.

DEC simulations usually combine a small break size with a major failure (i.e. SI automatic start…), or a more stringent initiator than DBA initiator. DEC studies use realistic hypothesis, no single failure, and can considers operator actions. Initial states are generally at full nominal power, some specific cases are at hot zero power.

As explained in D2.5 and also in above summary of D2.1, in order to calculate properly the radiological consequences, it is needed to consider:

- Thermal hydraulic (TH) phenomena
- Thermo-mechanical phenomena
- FP behavior (in fuel, primary system, containment)

D2.5 distinguishes 3 big types of methodologies (these are also illustrated by a below copied picture of the project grand agreement [1] in order to consider the above 3 aspects) from the most to the least developed:

1. Simulations of all aspects (TH, thermo-mechanic, and FP behavior) by independent dedicated codes or integral codes
2. Simulations of TH and thermo-mechanic by dedicated codes, but FP behavior conservatively estimated
3. Simulations with TH code, but thermo-mechanic and FP behavior conservatively estimated.

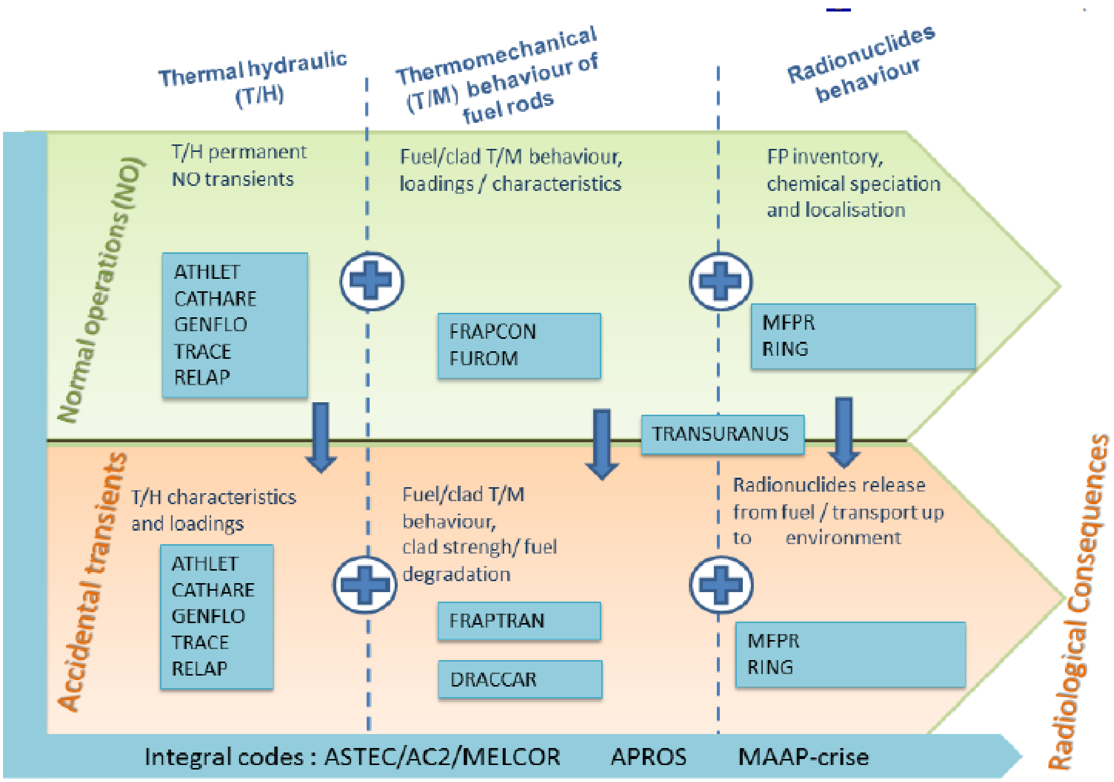


Figure 3 : Generic evaluation methodology main steps, associated phenomena and supporting codes

Table 2 of D2.5 [9] describes more in details which method has been followed by each partner, not reproduced here.

The §2.6.1 of D2.5 [9] summarized the results of the LOCA evaluations, sorted first by reactor type, then by accident type (DBA or DEC).

By reference to the above Table 3 of the partners reactor test cases, most interesting results features are the following:

- ARB didn't perform thermo-mechanical analysis and considers that 100% of the gap release goes to the gas phase in the containment (to be compared with D2.1 as summarized in above §3.1.2). ARB calculates huge integral activities released to environment (1.25E14 Bq at 80000 s for VVER-440 DBA case)
- EK performed a TH calculation concluding in a 900°C clad peak T for DBA. A thermo-mechanical analysis predicts no burst of the cladding, but anyway EK considers 100% of the gap content as released. In addition, FP from the fuel fragmentation is assumed to be released. Integral activity to environment is also huge (~9.5E13 Bq instantaneously released for VVER-440 DBA case)
- SSTC didn't perform thermo-mechanical analysis but took similar strategy as for EK (all the gap content is released). Result 2E13 Bq at 86400 s for VVER-1000 DBA case
- UJV has also a similar strategy as above EK and SSTC, but anyway leading to smaller releases ~1.6E11 Bq at 60000 s for VVER-1000 DBA case
- VTT presented for VVER DEC-A particular results compared to others. No containment model has been considered because no fuel damage has been expected. This was confirmed by the TH and thermo-mechanic calculations showing 890°C cladding peak surface T. Based on this, no release calculation has been produced. For EPR DBA, the worst calculations present damaged fuel rods but no release calculations have been provided
- ENEA provided a detailed TH and thermo mechanical analysis for PWR DBA case showing 860°C cladding temperature, what results in failure of 20 fuel assemblies out of 157. ~6E12 Bq to environment at 180000 s has been calculated
- IRSN provided a detailed TH and thermo mechanical analysis for PWR DBA case showing 750°C peak cladding temperature, what results in a failure of about one third of the fuel rods. FP behavior is also calculated. ~3.5E13 Bq to environment at 180000 s has been calculated
- HZDR provided a TH calculation of a double ended guillotine break (DBA case) for PWR, what led to 1050°C peak cladding T. Thermo mechanical analysis showed that 10% of core rods presents cladding rupture. No calculation of FP product has been provided. All the source term is supposed to be transported to the containment and the containment release rate is calculated for maximum containment pressure. No decay of FP is supposed. ~5.84E14 Bq to environment at 86400 s has been calculated
- LEI provided a TH calculation of a double ended guillotine break (DEC case) with delayed actuation of LPCI for BWR, what led to 830°C maximal peak cladding surface temperature. Thermo-mechanic calculation led to the failure of about 55% of the fuel rods. Path of fission products from containment to environment are considered through containment design leakage. ~5.3E14 Bq to environment at 200000 s has been calculated.

What can be observed is thus a quite big variability of the releases: 3 orders of magnitude difference between 1.6E11 Bq (VVER-1000 DBA case of UJV) and 1.25E14 Bq (VVER-440 DBA case of ARB) only partly explained by different scenarios considered and time period of releases considered. Remark that VTT presents no release for the VVER for the DEC-A case studied (because of no fuel damage).
For PWR-900, considering the same scenario and time period of releases, smaller differences were observed (6E12 to 3.5E13 for DBA case respectively from ENEA and IRSN) partly explained by difference in T/H calculation results and assumptions regarding containment leaks.

4.2 Overview of the initial SGTR simulation reports

All chapters dedicated to SGTR are the §2.2.2, §2.3.2 §2.5 and §2.6.2 of D2.5 [9].
See also the §2.2 of the present deliverable.

The SGTR TH simulation usually requires a model of the primary and secondary systems, with a break that creates a mass exchange. BWR are exempt from this accident because of no SG.

The DBA and DEC A scenario have been fixed by the task 2.2 of the project.

DBA simulations of PWR consider usually one single tube guillotine break (at various locations), and use conservative initial and boundary conditions, and of course a single failure. Moreover, SGTR is one exceptional example of DBA for which operator actions are needed during the first 30 minutes. Such actions must be simulated in the study with a selected timing.

DBA simulations of VVER are also considering a collector cover lift up.

DEC simulations of SGTR are considering several tube breaks, or the coincidence of SGTR with steam line break, or a SGTR with a major system failure. Remark that BOKU was considering a SGTR of one tube at the top of the tube bundle with a SGRV stuck open. Such scenario is considered as a mandatory DBA SGTR in Belgium (a blocked open SGRV being considered as a potential single failure for DBA studies in Belgium).

Table 3 of D2.5 [9] describes more in details which method has been followed by each partner, not reproduced here. As a remark, note that the primary source term is the spiking and has the advantage to be independent of the TH transient (contrarily to the LOCA) as already explained in §2.2 : there is thus usually no thermo-mechanical calculations to consider for the SGTR. For sure, this spiking model is a fundamental hypothesis of any SGTR study.

The §2.6.2 of D2.5 [9] summarized the results of the SGTR evaluations, sorted first by reactor type, then by accident type (DBA or DEC-A).
For VVER calculations (5 partners), the main features are the following. Firstly, a variability in the mass of liquid/steam which is discharged through the break from the primary to secondary system, and then from the affected SG to the environment: from 50 to 760 tons for DBA cases (60 to 850 for DEC-A cases) This variability is due to the different kinds of initiators, different release time periods, but also to other hypotheses like the SI duration.

Therefore, the calculated release to the environment varies from 3E12 Bq (at 4000 s) up to 2E15 Bq (at 22000 s) for DBA cases (1E13 Bq at 4250 s to 2.4 E14 Bq at 5000 s). The variability for DBA cases is similar compared to LOCA results and is explained not only by the variability in the mass of liquid/steam released but also by the variability of the source term (spiking) considered (also narrower boundaries compared to the LOCA).

For PWR results (5 partners) the mass of liquid/steam which is discharged through the break also greatly vary and then from the affected SG to the environment (from 75 to 190 tons for DBA cases and from 4 to 390 tons for DEC-A cases). Therefore, the calculated activity released into the environment varies from 2.4 E9 (at 1800 s) up to 4.7E13 Bq (at 3240 s) for DBA cases and from 1.2E9 Bq (at 3000 s) to 9E11 Bq (at 8000 s) for DEC-A cases. Note that the particularly low result of 1.2E9 Bq was obtained by CIEMAT PWR-1000 DEC-A study during which the release to the environment is due to a coincident SLB which can be isolated.

5 Discussions about model improvements

5.1 LOCA

As above show in §1 (figure 3: project structuration), the main objective of WP3 was to improve models for LOCA.

Among these improvements, a great deal of effort was devoted to better estimating and taking into account the number of failed fuel rods during a LOCA transient for the specific DBA and DEC-A conditions. It can indeed provide significant differences in source term prediction compared to current LOCA DBA safety analyses (shown in D2.1) where only thermal-hydraulics analyses (occasionally supported by decoupled radioactive transport codes) are performed, conservative assumptions are assumed, and decoupling factors are used for the failed fuel rod fraction and then for the source term evaluation.

5.1.1 Primary circuit source term: fuel rod burst failure and gap inventory release

Depending on the countries and their current safety regulations/rules the fixed number of failed rods in DBA vary from 10% to 100% for a large break LOCA transient [3]. Only few methodologies were really suited to evaluate the number of failed rods. This is due to several limitations in the modelling of the complex and coupled thermohydraulic and thermomechanical phenomena occurring during a LOCA and leading to the fuel clad ballooning and burst.
During the project, in order to better predict the clad ballooning and burst in LOCA DBA and DEC-A conditions the following work has been done:

- Cladding creep models were updated in some codes. Especially a best-estimate fitting of the parameters of the plastic deformation for E110 alloys was performed and introduced in one of the used code.
- For Zr-4 alloys new and various clad burst criteria (based on temperature and stress and strain) were established since most of the available ones were developed to predict flow blockage for reflooding evaluations and were not suitable for DBA and DEC-A conditions with smaller clad deformations. Lower and upper engineering stress envelopes were also defined allowing an estimation of the range of burst fuel rod fractions. A specific work was also conducted on E110 alloys allowing to establish a best estimate burst strain limit and derive a conservative burst strain limit.
  All this work has first consisted in revisiting available experimental databases: mainly Zr-4 clad burst tests (performed within different experimental conditions) with advanced scanning methods and Russian E100 clad burst and ballooning tests re-investigating in more details their geometry with more accurate measurement techniques (such as the computer tomography scan). Using these new criteria and models, burst parameters of the experimental tests included in the different databases were re-assessed.
  The re-assessed burst temperatures of Zr-4 alloys were generally found to be more consistent with the experimental ones while the burst strains were always underestimated. The new established criteria implemented in some codes were thus found to be suitable to give a best-estimate (somewhat conservative in some cases) estimates of the number of failed rods and thus on the associated radiological consequences. For E110 alloys, the updated plastic deformation parameters together with the use of the conservative burst strain limits gave reliably conservative predictions.
- Refined full core modellings were developed to more accurately capture the fuel thermohydraulic and thermomechanical behaviour fuel rod by fuel rod or a least fuel assembly by fuel assembly. In particular full 3D core modellings were developed by some partners which are more able to capture distinctive fuel assembly behaviour and therefore better predict the rod burst ratio according to their distribution within the core and their characteristics of rod (i.e. power, burn-up, internal pressure…). This work was very challenging, since this refined modelling was generally very computational-time consuming and challenge the current computational calculation limits. Therefore, in some cases, when the computational time exceeded the computational calculation limits some compromises were found by partners and/or an adapted core nodalisation used. Two kinds of approaches were especially used with:
  - A 3D modelling of the Reactor Pressure Vessel able to capture the asymmetric response of the core during the LOCA associated to the heterogeneous flow distribution in the primary loops and where each fuel assembly was modelled separately. Four representative fuel rods per channel were modelled. Different burst models can be used, the simplest one being the maximum strain criterion, but only one single value for the initial rod internal pressure. This approach, though very promising, required very long computing time.
  - A 3D core thermalhydraulic model of an eighth to a fourth of the core where in each core channel the corresponding fuel assembly was described by some weighted structure and where the 2D thermomechanical behaviour of the fuel rod object was evaluated. This approach using the coupling of thermohydraulics and thermomechanics to calculate an average thermalhydraulics at fuel, assembly scale and predict the corresponding behavior of the fuel rod object, was found to be the best compromise to depict the core heterogeneities in fuel assembly characteristic distribution while keeping the computational calculation time under the limits.

Finally, a more refined modelling of fuel gap inventory was also performed by considering a 3D distribution of fuel burn-up. Improvements of FP release models from fuel were performed within the project especially in the dedicated FP codes describing their behaviour at grain scale. Dedicated effort was done on the specific enhanced releases of fission gases from the high burn-up structure at the fuel pellet periphery. These FP code improvements greatly benefited from their coupling with fuel performance codes that has also been upgraded during the project by considering the thermomechanical effect on FP releases. These improved coupled tools were however not used for reactor applications during the project but instead constant FP release values with respect either to the initial or gap inventory were used depending on the FP category (i.e. noble gases, volatiles FP such as Cs & I, semi-volatiles…). Depending on partners either best-estimate averaged values or very conservative approaches were used.
While the number of failed fuel rods has a significant role on the amount of FP released in the containment gas phase and then on the radiological consequences of a LOCA, other phenomena such as the FP release rates from fuel are equally important. Within the R2CA project the first has been widely investigated and considered in the updated reactor calculations while the second aspect though also widely investigated and improved was not applied.

5.1.2 Transfer of the contamination and FP behaviour in containment

No specific improvements or developments of model for FP transport in the primary circuit was performed within the project but a re-assessment of the models implemented in some codes was performed where these models were generally developed/validated for severe accident conditions (i.e. for higher temperatures, FP concentration…). The behaviour of iodine was especially focused on as the R2CA experimental database includes several experimental dedicated tests investigating the effect of chemical reaction kinetics on iodine speciation in the primary circuit. In general, a good agreement between experimental data and calculations were observed.
Finally, few improvements were performed regarding FP behaviour in the containment. It mainly concerned the gaseous iodine interaction with dry paints where the adsorption/desorption parameters used in a dedicated simulation tool were fitted to the Ameron Amerlock paints data, for VVER reactor application.

5.2 SGTR

As above show in §1 (figure 3: project structuration), the main objective of WP4 was to improve models for SGTR.

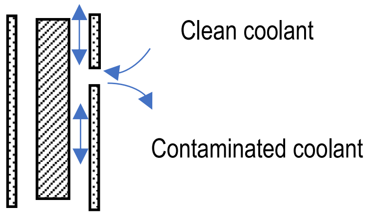
Among these improvements, the refinement of the spiking model appears to be feasible without unduly costs and can provide a significant added value to a SGTR calculation of releases to environment whatever the scenario (overflowing or non-overflowing). Improvements were made during the project, for the most part based on legacy or new NPP measurements.

5.2.1 Primary circuit source term

a) *Leaks through defects in cladding in normal operation*

The presence of leaking fuel rods prior to the accident are the basement of the considered source term for a SGTR (tramp uranium is also a potential source but at a lower level). By comparison, such hypothesis doesn't appear for a LOCA for which the damage to the fuel is supposed to happen as a consequence of the accident itself.

Leaking fuel rods are fuel rods having a defect in their cladding, leading to the presence of coolant in the space between the cladding and the pellet. Due to the nuclear reaction, salts containing fission products like iodine are dissolved in the coolant which is present in this space. Due to the different nature of the clad default compared to the LOCA, gasses can here accumulate in the upper part of the fuel rod and are therefore not released, but such feature is not systematic (i.e gasses can be also released by the fuel rods during SGTR). In stationary operation, an equilibrium takes place between the (clean) coolant entering in the rod through the defect, and the contaminated water going out of such fuel rod ([83] page 126).



In case of presence of such leaking fuels rods, the CVCS filters the primary coolant such as to limit the contamination in the primary.
The presence of these defects is thus detected by the measurement of the specific activity of different radioactive species in the primary system. It's important to remark that several orders of magnitudes of such measured specific activities are observed between situations without or with leaking fuel elements.
Moreover, maximal authorized values of the specific activities for several given species in the technical specifications are fixing boundary situations, and the plant is thus obliged to stop (and to identify and discharge leaking fuel assemblies) in case of non-respect of these limits.
Such limiting situation at the limit of the technical specifications can be postulated as initial situation for the safety demonstration of the NPP's licensing, like for the SGTR.

For the current project, partners used their own specific values of initial activities, based on maximal value as specified in the technical specifications, or based on NPP measurements/feedbacks, as described in §4.1 of D2.1 [2].

b) Spiking phenomenon

Starting from a stable situation at power with leaking fuel elements, a transient like a scram can initiate the expulsion of a big quantity of contaminated water out of the fuel rod. Such phenomenon is called the spiking.
The intensity (a part or all the water inside the gap) and the time constant of such spiking depend on several complex factors: numbers of leaking fuel rods, type and position of cladding defects, fuel burn-up, type of transient…

In the past, like in [80] (similar project of the EC in 1995), basic spiking models were using a simple multiplicative factor.
The release during the spiking was R (Bq/s) = k R_0 with k being the spiking factor, and R_0 being the release in stationary conditions. In [80], participants were using k coefficients from 30 to 100. This model was simple but very conservative, and the release duration must be fixed otherwise the total source term becomes infinite.

Indeed, in reality, once released in the primary coolant during such spiking, the iodine inventory can be restored but this requires a period of irradiation which is much bigger than the duration of a SGTR event. As a fundamental consequence, the source term representing such spiking phenomenon during a SGTR event is limited.
Iodine is usually the most representative species considered with different radioactive isotopes that can be either considered separately or represented by I-131 only with a conversion factor.

As this is performed in Belgium (Bel V/TE) or in France (IRSN) using a spiking model with a time constant, leads to a variable primary activity during the SGTR event, which is a complexity because this evolution has to be calculated (TH code and explicit model) : R (Bq/s) = Inv/τ * exp (-t/τ) with "Inv" being the total source term to be released during the phenomenon and τ the time constant of the release.

During the project, regarding "iodine" spike phenomenon few partners have performed some modelling improvements (i.e. in RING code….) or developed new models in the codes they used (i.e. in MELCOR code.…) providing a time dependence evolution of the activity.

Going further in the search of a more realistic spiking model, deliverable [7] (T4.2 of the project) describes how the RING codes calculates the spiking during the SGTR event, based on an empirical model that considers cladding failures and tramp uranium. This model considers the effect of several parameters such as pressure and power reduction during the SGTR event, as well as boric acid concentration evolution. Such model is calibrated based both on steady-state situation and on measurements during real spiking transients on site. This approach is very valuable because can lead without unduly costs to significant reduction of SGTR radiological consequences, staying at the same time conservative but closer to the reality.

Note that these cladding defects can have different kinds of origin like the fuel fabrication process, the cladding design, fretting or the presence of foreign objects in the primary system. Without going into details, significant improvements have been performed in the history: the presence of at least one leaking fuel assembly was a common situation for any cycle in the past while became today rarer ([84]). As example, since several years, not any leaking fuel assembly has been detected in any belgian NPP. For the same reason, maximal authorized values of technical specifications have been reduced in the history of belgian NPP's, what led to significant reductions of the official SGTR licensing studies.

5.2.2 Transfer of the contamination through the SG break

This aspect is much more complex than for the LOCA for which all primary coolant content is usually supposed to be released to the containment. See figure 2 in above §2.2.

For the SGTR, there are several aspects to consider (see figure 2 in previous §2.2):

- Firstly, the SG break flow and its specific activity can be both time dependent. A precise evaluation of the break flow during the SGTR requires a TH calculation code and a model of the NPP.
- Secondly, in case of uncovered breaks, the break flow can split in 2 phases due to thermodynamic flashing. Each phase will have its own specific activity in iodine (or other volatile species) taking into account the partitioning which depends on the chemical form of the considered species (i.e. iodine or other volatile species).
- Thirdly, in case of uncovered break, a part of the liquid phase can be transformed into small droplets (atomisation) that can have their own dynamic feature, compared to the other part that will mix with the secondary water.
- Fourthly, partitioning from the secondary as well as carry-over can further occur.

For the first aspect, such evaluation of the break flow is strongly connected to the produced deliverables of the present project regarding the TH simulation of SGTR (see below §4) and depend on the reactor model, the scenario (location of the break…) and transient (HPSI injection…). Regarding this aspect, the TH code simulations were generally quite detailed, and no improvements/modifications were then deemed necessary.

For the second and third aspects, the hypothesis used by some of the partners for the FP mass transfer and the distribution of the iodine chemical species in the liquid and gas phase in the affected SG were respectively discussed in §4.3 of D2.1 and displayed in Figure 11 of D2.1. However, during the project few partners have performed improvements in their modelling (i.e. in ASTEC considering the iodine flashing rate dependency on chemical speciation…) or have developed new functionalities in the codes (i.e. in MELCOR.…) they used to better take into account these two aspects.

Finally, regarding the fourth aspect, improvements were also performed in few codes for the iodine partitioning model with the consideration of the dependence of the partition coefficient on the SG liquid temperature and the implementation of a new model for the liquid-gas mass transfer (based on the two-film theory) speeding up the mass transfer kinetics from the liquid to the gas phase in SG evaporative conditions. Applied to molecular iodine these modifications led in average to a higher partitioning coefficient then a higher maximum molecular iodine activity in the gas phase) and a much higher contribution of the partitioning phenomenon in the environmental releases compared to the initial calculations (40% vs 1%) where a constant partitioning coefficient of 100 was considered.

5.2.3 Transfer of the contamination from the affected SG to environment

This last step of the calculation can be compared with the previous §3.1.2 regarding LOCA (transfer from containment to environment).
For the SGTR transients, as mentioned in §4.4 of D.2.1 [2], and as considered by the partners, most important part of the release to environment can take place through the secondary discharge valves. These valves can release contaminated steam or even, contaminated water if the affected SG is overfilled during the transient.
A part of the activity will remain trapped in the secondary system, mostly in liquid water of the affected SG.
The retention in the upper SG structures is however not considered by the partners.

Therefore, as for the previous step, all these aspects (opening timing and the flows through relief valves, quantity of liquid water remaining on affected SG) can be either matter of simplified hypotheses or matter of explicit TH calculations using a TH code and required model. Some improvements in the discharge flow modelling have been made by some partners in their TH codes.

6 Second set of studies using improved models for RC calculations

As for the first set of calculations, technical individual reports have been issued by each partner including performed simulations, hypotheses, and calculation results. These reports are part of the task T2.5 “Reassessment of reactor test cases and quantification of gains” and are summarized in deliverable D2.7 [10.]
The present chapter is the image of the chapter 4 but reflects updated calculations using improvements/modifications.

The table 2 below describes all performed updated simulations, based on table 3.

	PWR LOCA	PWR SGTR	EPR LOCA	EPR SGTR	VVER LOCA	VVER SGTR	BWR LOCA
ARB					[47] [48] [49] [50]	[51] [52] [53] [54]	
BEL V		[55] [56]					
BOKU		[63][64]				[65][66][67]	
CIEMAT		[78][79]					
EK					[57]	[58]	
ENEA	[59][60]						
HZDR	[77]						
IRSN	[61]	[62]					
LEI							[68]
SSTC					[69][70]	[71][72]	
TRACTEBEL (TE)		[73]					
UJV					[74]	[75]	
VTT			[76]				

Table 2: Partners updated reactor test cases

6.1 Overview of the updated LOCA simulation reports

See §3.1.1 of D2.7 [10]. The remarked improvements of the partners calculations are the following:

- ARB refined his initial very conservative assumption of the first calculation set (100% failed rods) and also the transfer from primary to containment. BORON model of ATHLET was used.
- EK used a new cladding failure criterion (original model was also very conservative) in FRAPTRAN and refitted the plastic deformation model.
- SSTC refined his RELAP5 model of the core, in order to obtain better local TH conditions, themselves transmitted to TRANSURANUS to calculate the cladding behavior.
- UJV improved the dry paint deposition model in the containment refitting the model parameters for AMERLOCK paints used in VVERs. The source term to containment is postulated independently of the TH calculation.
- IRSN made improvements in different parts of the calculation chain, using new simulation tools (DRACCAR allowing a 3D core description instead of ICARE) and models (new fuel clad burst model/criteria).
- ENEA used the same tools but with a refined model of the core. Collaboration with IRSN to use new fuel clad burst model/criteria.
- HZDR used a new core model combining 3D thermal-hydraulic model of the RPV with the fuel rod thermomechanical model.
- VTT used new simulation codes in their calculation chain replacing CASMO by SERPENT. Also, a new fuel clad creep and a new clad burst model (in collaboration with IRSN) were used.
- LEI used the same calculation chain. Additional TRANSURANUS calculations were performed to refine the BWR core model.

6.2 Overview of the updated SGTR simulation reports

See §3.1.2 of D2.7 [10]. The remarked improvements of the partners calculations are the following:

- ARB refined their calculation in similar way as for their LOCA calculations, using BORON model of ATHLET. Also drift flux model was improved to better simulate discharge through SG relief valve.
- Bel V used the CATHARE radioelement transport model to assess the I-131 release to the environment rather than using simplified model and hypothesis.
- EK improved their spiking model using RING code for I & Cs (see the previous §3.2.2 in the present deliverable about this subject). Initial primary coolant activities for other species than iodine or cesium are also better taken into account based on new NPP feedbacks.
- SSTC refined their boron concentration model in RELAP to represent source term transfer in primary and secondary circuits.
- UJV improved their activity transport model, using a new methodology & a computational analytical approach representing 140 isotopes.
- BOKU developed their own spiking model based on empirical data. Clean-up and pool scrubbing effect were also included through specific post-processing functions in RELAP5 code.
- IRSN used a new calculation chain with new modules for activity transport in primary (including the spiking effect) and for activity transfer to secondary (including jet flashing and atomization). The thermal fragmentation of the jet at the break location is also modelled. FP speciation in the primary circuit is characterized and the iodine speciation is especially considered through different flashing rates considered at the break.
- CIEMAT developed two external functions in MELCOR for iodine spike modelling (forced-convective release model driven by temperature and pressure changes) and for activity transfer from primary to secondary through the break of failed SG (including flashing, atomization and partitioning).
- TE improved the iodine partitioning model by considering the temperature effect and the evaporative conditions. New EOPs were also considered.

7 Conclusions and recommendations sorted by subject

7.1 Preliminary remarks and boundaries of recommendations

As a reminder, the objective of the project was to reduce the degree of conservatism in safety evaluations of LOCA and SGTR bounding scenarios within DBA and DEC-A conditions by improving the simulation tools and the calculation methodology.
Once arrived at the end of the project, as shown in the previous chapters, there is a large amount of provided calculations by the partners: first set and second set of calculations with proposed improvements. Quantitatively, as also shown in previous chapters and in D2.7 [10], an important dispersion appears in terms of releases values (orders of magnitude) in both sets of calculations. This is the result of several factors like the hardware (PWR, VVER, BWR), the used calculation tools and methodologies, the duration of the release period considered, the list of isotopes and selected improvements between the 2 sets of calculations.
Despite these differences between them, these calculations can be far from rough conservative evaluation, as already shown in the deliverable D.2.1 [2] early in the project and therefore contribute to a better estimation of the safety margins.
In some cases, partners did not find useful to perform any calculation, or to perform a second set of calculation. Different interpretations of such situations can exist such as that the existing safety demonstration is sufficient without performing more realistic calculations, or that further improvements are judged marginal and not valuable to perform. In any case, this is a factual observation of the project R2CA to notice.

The following subject are not part of expressed recommendations by the project:

1. The calculations that were performed during the project deal with radiological releases to environment. A simplified tool was used to calculate the different doses (thyroid & total effective dose) just to quantify the gains in terms of RC between the first & second set of calculations. The aim of the project doesn't encompass discussions about methods to calculate doses. This is consistent with part B of grant agreement §1.3 [1].
2. The choice of conservative initial and boundary conditions of DBA studies like LOCA and SGTR are matter of the regulation texts or agreements between the safety authorities and the Utility, in order to sustain the safety demonstration of a given NPP. These rules are often "country dependent" and therefore not discussed in the framework of R2CA. The DEC-A studies are less affected by these rules as best-estimate conditions are usually tolerated.
3. Specific hardware characteristics or hardware improvements on installations, or on fuel fabrication, clearly impacts the result of radiological releases. The spirit of the part B of grant agreement §1.3 [1] is to start from "a fixed situation" in order to perform the analysis, so not to discuss about any hardware improvement. Nevertheless, optimization of EOP's were matter of analysis in different deliverables of the project, and also part of improvements notified by some partners in the second set of calculations.

7.2 Recommendations

The below recommendations are sorted by accident type and by the capacity of a barrier to retain a part of the FPs up to the environment.

7.2.1 Recommendations for LOCA

Regarding the fuel release, it is recommended:

- to refine the FP's gap inventory by considering differences between fuel rods/assemblies (multi-inventory, 3D burn-up distribution)

- to have a coupling between thermo-mechanical and TH calculation, using a 3D core model, in order to better predict the number of damaged fuel rods during the accident. This need is reinforced by the asymmetric character of the LOCA accident. The level of details of the model is to be balanced with CPU capabilities (compromise that still evolves with CPU capabilities evolution).
- to express specific rod initial conditions (burnup, internal pressure, power) for thermo-mechanic calculations (including axial gas communication).
- to use refined criteria of "cladding burst temperature and strain", specific to DBA & DEC-A conditions as developed during the project.

Regarding the behaviour in the containment and the release to environment, it is recommended:
- to distinguish different groups of FPs with similar behaviour (i.e. noble gases, aerosols, iodine….) –
- to use a dedicated calculation tool/module for modelling the iodine behaviour in the containment. Speciation of iodine in the containment is indeed rather complex, time dependent, and impossible to evaluate without a dedicated calculation code.
- to consider the retention of FPs on the surfaces and for iodine especially, considering the difference in its retention on paints as a function of the nature of paint, which differs from one NPP concept to another.

In addition, the use of:
- detailed fuel performance code can be beneficial for core model refinement through a better characterisation of the fuel rod/assembly behaviour during the transient
- coupled fuel performance and FP release codes can be beneficial for a better estimation of the FP releases during the transient (such as a potential increased release of FPs due to fuel oxidation, stress variations, …) and/or quantifying the FP releases from specific fuel zones (i.e. from the High Burn-Up fuel structure…).

7.2.2 Recommendations for SGTR

For the SGTR, as already mentioned, the source term (release through fuel rods) doesn't depend as much as for the LOCA on the thermal hydraulic transient, and this constitutes an important difference compared to the LOCA. Moreover, due to the different nature of the clad default compared to clad burst, a part of the noble gases can be considered as remaining trapped inside the fuel rods, but this is not systematic (i.e. an amount of noble gasses can be also released during SGTR).

As a general recommendation for SGTR, EOP's screening and optimization is identified as potential improvement.

Regarding the fuel release and transport to primary, it is recommended:
- to use best-fitted spiking correlations based on experience and/or NPP measurements. Among them, some are developed during the current project (ex : in RING code)
- if available, to use more mechanistic codes that are able to model the spiking (i.e. coupled fuel performance and FP release codes such as TU/MFPR-F..). Indeed, improving these spiking models still remains a valuable subject of development for the future
- to consider filtration and dilution in the primary system

Regarding the transport from primary to secondary system, it is recommended:
- to screen initial and boundary conditions in order to distinguish scenario of flooded break versus uncovered break
- when possible, to evaluate the speciation of iodine in primary circuit conditions (i.e. as done during the project using the SOPHAEROS module of ASTEC) in order to determine the volatility that could impact flashing fraction at the break in case of uncover break and its partitioning for flooded break
- when possible, using codes that are able to model the aerosol production (called atomization and/or jet fragmentation) at the break (i.e. as done during the project using the DROPLET module of ASTEC)

- to use test facility results (i.e. from the ARTIST experimental program) to determine the aerosol retention inside the secondary circuit in the upper SG structure such as the swirl-lane or the chevron separators) as a function of their sizes.

Regarding the transfer from secondary system to environment, it is recommended:

- to consider specific evolution of activities in liquid and gas phases of the secondary system (i.e. as done using the SAFARI module of ASTEC)
- to examine models of SG relief and safety valves and refine them when needed

In addition, due to the potential risk of embrittlement and rupture of defective fuel rod clads after their secondary hydriding (potentially significantly increasing the source term in primary circuit due to partial fuel dissemination…), it is recommended to use dedicated simulation tools able to simulate the overall phenomena taking place during the secondary hydriding of the internal surface of the defective fuel rod clad (i.e. from hydrogen uptake up to defect (blister/sunburst) formation).

8 References

8.1 Main used deliverables of the present project for D2.8

1. GRANT AGREEMENT NUMBER 847656 — R2CA
2. Review of the RC evaluation methodologies, Final report, Deliverable D2.1, IRSN, May 2020
3. Report on Fuel Fragmentation, Relocation and Dispersal, NEA/CSNI/R(2016)16, 2016
4. D3.4 Rod cladding failure during LOCA - Final report on experimental database reassessment and model/code improvements (IRSN), Version 01, February 2023
5. D3.6 Fuel rod behaviour during LOCA transient – Final report (JRC), Version 01, December 2022
6. D4.2 Final report on experimental database reassessment and on model/code improvements for fission product releases during a SGTR transient (CIEMAT), Version 01, July 2023
7. D4.4 Final report on fission product release from defective fuel rod during a SGTR transient (POLIMI), Version 01, February 2023
8. Improvement of the iodine and ceasium spiking models in the RING code, T2.4, EK, May 2021
9. D2.5, Reactor test case simulations, Version 04, LEI, December 2021
10. D2.7, Reassessment of reactor test cases, version 1, LEI, October 2023
11. Presentations during the final project meeting and the final workshop - IRSN - November 2023

8.2 List of all reactor test case simulations: task T2.3

12. LOCA DBA of VVER-440, Version 01, ARB, March 2021
13. LOCA DEC-A of VVER-440, Version 01, ARB, March 2021
14. LOCA DBA of VVER-1000, Version 01, ARB, March 2021
15. LOCA DEC-A of VVER-1000, Version 01, ARB, March 2021
16. SGTR DBA of VVER-440, Version 01, ARB, March 2021
17. SGTR DEC-A of VVER-440, Version 01, ARB, March 2021
18. SGTR DBA of VVER-1000, Version 01, ARB, March 2021
19. SGTR DEC-A of VVER-1000, Version 01, ARB, March 2021
20. SGTR analysis of a generic PWR 1000 MW - DBA scenario, Version 01, Bel V, March 2021
21. SGTR analysis of a generic PWR 1000 MW - DEC-A scenario, Version 01, Bel V, March 2021
22. Reactor test case simulation-CIEMAT SGTR-DBA simulations, Version 0, CIEMAT, March 2021
23. Reactor test case simulation-CIEMAT SGTR-DEC-A simulations, Version 01, CIEMAT, April 2021
24. DBA LOCA analysis of a VVER-440 reactor, Version 01, EK, February 2021
25. SGTR analysis of a VVER-440 NPP, Version 01, EK, February 2021
26. LOCA DBA analysis of a generic PWR 900 MW, Version 01, ENEA, March 2021
27. LOCA DEC-A analysis of a generic PWR 900 MW, Version 01, ENEA, March 2021
28. LB-LOCA analysis of Konvoi PWR (DBA and DEC A), Version 01, HZDR, March 2021
29. First set of LOCA DBA calculations, Version 01, IRSN, May 2021
30. First set of LOCA DEC-A calculations, Version 01, IRSN, May 2021
31. SGTR analysis of PWR, Version 01, IRSN, July 2021
32. Generic PWR-DBA scenario(SGTR), Version 01, LEI, March 2021
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34. SGTR analysis of VVER DBA scenario, Version 02, LEI, October 2021
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36. LOCA analysis of BWR-4 MARK I, Version 01, LEI, February 2021
37. LOCA DBA analysis of VVER-1000/320, Version 01, SSTC, February 2021
38. R2CA-Task 2.3 : template for LOCA DEC-A, SSTC
39. SGTR DBA analysis of VVER-1000/320, Version 01, SSTC, February 2021
40. SGTR DEC analysis of VVER-1000/320, Version 01, SSTC, February 2021
41. Reactor test case simulation - Tractebel-SGTR simulations-DBA, Version 00, Tractebel, February 2021
42. Reactor test case simulation - Tractebel-SGTR simulations-DEC-A, Version 00, Tractebel, March 2021
43. Summary report on VVER 1000/V320 LOCA calculations, Version 02, UJV, February 2021
44. Summary report on VVER 1000/V320 SGTR calculations, Version 01, UJV, March 2021

45. R2CA-Task 2.3 : Template for LOCA, VTT
46. LBLOCA analysis of VVER 1000/320 (Kozloduy NPP unit 6) with Apros and FRAPTRAN-GENFLO, Version 02, VTT, February 2021

8.5 List of all updated reactor test case simulations: task T2.5

47. DBA LOCA analysis of VVER-440, Version 01, ARB, June 2023
48. DEC-A LOCA analysis of VVER-440, Version 01, ARB, June 2023
49. DBA LOCA analysis of VVER-1000, Version 01, ARB, May 2023
50. DEC-A LOCA analysis of VVER-1000, Version 01, ARB, May 2023
51. DBA SGTR analysis of VVER-440, Version 01, ARB, March 2023
52. DEC-A SGTR analysis of VVER-440, Version 01, ARB, March 2023
53. SGTR analysis of VVER-1000 DBA, Version 01, ARB, February 2023
54. SGTR analysis of VVER-1000 DEC-A, Version 01, ARB, February 2023
55. SGTR analysis of DBA case, Version 01, Bel V, October 2023
56. SGTR analysis of DEC-A case, Version 01, Bel V, October 2023
57. DBA LOCA analysis of a VVER-440 reactor with the improved clad burst model, Version 01, EK, December 2022
58. Reassessment of activity release during SGTR in a VVER-440 NPP, Version 01, EK, October 2022
59. LOCA DBA analysis of a generic PWR 900 MW, Version 01, ENEA, December 2022
60. LOCA DEC-A analysis of a generic PWR 900 MW, Version 01, ENEA, December 2022
61. Second set of LOCA DBA & DEC-A calculations, Version 01, IRSN, May 2023
62. SGTR DBA calculations in PWR 900 : additional scenario (DBA3), IRSN, September 2023
63. DBA SGTR analysis of generic PWR, Version 01, LEI, December 2022
64. DEC-A SGTR analysis of generic PWR, Version 01, LEI, December 2022
65. DBA SGTR analysis of VVER 1000, Version 01, LEI, December 2022
66. DEC-A SGTR 1 analysis of VVER 1000, Version 01, LEI, December 2022
67. DEC-A SGTR + MSLB analysis of VVER 1000, Version 01, LEI, December 2022
68. LOCA analysis of BWR-4 MARK I, Version 01, LEI, December 2022
69. LOCA analysis of DBA case, Version 01, SSTC, September 2022
70. LOCA analysis of DEC-A case, Version 01, SSTC, September 2022
71. SGTR analysis of DBA case, Version 01, SSTC, August 2022
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76. RE-simulations of EPR LB-LOCA with FRAPTRAN-GENFLO, Version 01, VTT, December 2022
77. DBA LB-LOCA analysis of Konvoi PWR with detailed core model, Version 01, HZDR, October 2023
78. Reassessment of reactor test case simulation – CIEMAT SGTR DBA simulations, Version 01, CIEMAT, July 2023
79. Reassessment of reactor test case simulation – CIEMAT SGTR DEC-A simulations, Version 01, CIEMAT, July 2023

8.7 List of references outside of the project

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