



HAL
open science

Review of the RC evaluation methodologies

Dorel Obada, Tadas Kaliatka, Paul Bradt, Simon Verdebout, Ruslan Lishchuk, Andrii Bereznyi, Dmytro Gumenyuk, Yuri Ovdienko, Asko Arkoma

► To cite this version:

Dorel Obada, Tadas Kaliatka, Paul Bradt, Simon Verdebout, Ruslan Lishchuk, et al.. Review of the RC evaluation methodologies. European Commission. 2020. irsn-04106723v2

HAL Id: irsn-04106723

<https://irsn.hal.science/irsn-04106723v2>

Submitted on 10 Aug 2023

HAL is a multi-disciplinary open access archive for the deposit and dissemination of scientific research documents, whether they are published or not. The documents may come from teaching and research institutions in France or abroad, or from public or private research centers.

L'archive ouverte pluridisciplinaire **HAL**, est destinée au dépôt et à la diffusion de documents scientifiques de niveau recherche, publiés ou non, émanant des établissements d'enseignement et de recherche français ou étrangers, des laboratoires publics ou privés.



Distributed under a Creative Commons Attribution 4.0 International License



European
Commission

Horizon 2020
European Union funding
for Research & Innovation



**REDUCTION OF
RADIOLOGICAL
ACCIDENT
CONSEQUENCES**

Action	Research and Innovation Action NFRP-2018-1
Grant Agreement #	847656
Project name	Reduction of Radiological Consequences of design basis and design extension Accidents
Project Acronym	R2CA
Project start date	01.09.2019
Deliverable #	D2.1
Title	Review of the RC evaluation methodologies
Author(s)	Dorel Obada (IRSN), Tadas Kaliatka (LEI), Paul Bradt (Tractebel), Simon Verdebout (Tractebel), Ruslan Lishchuk (ARB NPPS), Andrii Berezhnyi (ARB NPPS), Dmytro Gumenyuk (SSTC NRS), Yuri Ovdienko (SSTC NRS), Asko Arkoma (VTT)
Version	01
Related WP	WP2 METHO - Methodologies
Related Task	T2.1.1. Review of the release evaluation methodologies (IRSN)
Lead organization	IRSN
Submission date	14.05.2020
Dissemination level	PU



This project has received funding from the Euratom research and training programme 2014-2018 under the grant agreement n° 847656



History

Date	Submitted by	Reviewed by	Version (Notes)
09.10.2020	D. Obada (IRSN)	P. Bradt (Tractebel) N. Girault (IRSN)	01

Contents

1.	Introduction.....	5
2.	Workpackage WP2: METHOD – Methodologies.....	5
2.1	Objectives of the Workpackage WP2.....	5
2.2	Task 2.1.1 Review of release evaluation methodologies.....	6
2.2.1	Task 2.1.1 Objective.....	6
2.2.2	Context of each contribution.....	6
3.	Source term evaluation methodology in LOCA conditions	8
3.1	Isotopic inventory.....	8
3.2	Elements volatility.....	9
3.3	Fuel release.....	10
3.4	Release in the containment.....	11
3.4.1	Summary.....	12
3.5	Fission product behaviour in the containment	14
3.5.1	Fission product chemistry.....	14
3.5.2	Containment Spray System impact	14
3.5.3	Decay and filiation.....	16
3.6	Containment releases.....	16
3.6.1	Unfiltered releases	16
3.6.2	Filtered releases.....	20
4.	Source term methodology in SGTR conditions	21
4.1	Primary loop activity	21
4.2	Iodine form in the primary loop	22
4.3	Transfer in the Steam Generator	23
4.4	Secondary loop retention.....	25
5.	Summary and conclusions	26
	References.....	28
	Appendix A. Source Term evaluation methodology template	29



Abbreviations

ASTEC	Accident Source Term Evaluation Code
BDBA	Beyond DBA
BWR	Boiling Water Reactor
CSS	Containment spray system
DBA	Design Basis Accident
EPR	Evolutionary Power Reactor
FDNP	Fukushima Daiichi Nuclear Power plant
FP	Fission Product
LOCA	Loss-Of-Coolant Accident
MER	Modèle d'Evaluation de Rejets (Model for evaluating the releases) code
PWR	Pressurized Water Reactor
RC	Radiological Consequences
RCS	Reactor Coolant System
SA	Severe Accident
SGTR	Steam Generator Tube Rupture
ST	Source Term
VVER	Water-Water Energy Reactor?

1. Introduction

The R&D efforts on the evaluation of Severe Accident (SA) progression and consequences have been strengthened after the Fukushima Daiichi Nuclear Power plant (FDNP) accidents notably under research programs funded by the European Commission through the FP7 and the H2020 frameworks and by OECD/NEA/CSNI. These programmes led or are targeting to increase the predictability of SA progression, to assess Severe Accident Management (SAM) strategies or to improve mitigation of accident consequences. They also address, as a feedback of FDNP situations management, the reinforcement of SA simulation codes for Emergency Preparedness and Response (EP&R) diagnosis and prognosis activities. These consolidated evaluations of SA progression and consequences on the one hand and the potential improvements of SAM and mitigations strategies and devices on the other hand have been integrated into level 2 Probabilistic Safety Assessments (PSA2). The post-processing of these PSA2 results into classical risk diagrams (radiological consequences vs. frequency of occurrence) so demonstrates the effective or potential reduction of the risks associated to all main categories of SA conditions. This reduction so leads to lower down relatively the differences between the risks associated to Design Basis Accidents (DBA) and Beyond DBA (BDBA) conditions and confirms that the evaluations of radiological consequences for DBA conditions are done using very conservative deterministic assumptions mostly based on decoupled approaches. Those conservative assumptions and tools do not allow quantifying explicitly what would be the gains, notably in terms of RC, of additional safety measures or devices, improved accident management strategies and advanced fuel design.

To address this major issue, the project aims at consolidating assessments of RC of explicit DBA and DEC-A reactor accidental situations, focussing for simplicity sake on two main categories of accident: the Loss of Coolant Accidents (LOCA) and the Steam Generator Tube Rupture (SGTR) accidents, and at proposing the improvement of NPP accident management strategies and devices to reduce RC of these accidents for GEN II, III and III+ NPPs under operation or foreseen in Europe. The reactor cases will be mostly generic PWR and BWR where realistic VVER applications will be also performed to reinforce the demonstration of the operational capabilities of the approaches.

2. Workpackage WP2: METHOD – Methodologies

2.1 Objectives of the Workpackage WP2

This WP will concern the evaluation and improvement of methodologies that will support the demonstration of the approach. It will start by a comparative review of the methodologies implemented worldwide and reviews of both existing simulation codes and supporting experimental and analytical databases. Then evaluation tools will be used to simulate reactor for generic PWR and BWR configurations as analytical cases and realistic VVER and EPR. The results of these reactor cases will be analysed to confirm the orientation for the improvements of the codes for the two representative categories of accidents LOCA and SGTR. Then, during the last year of the project, the same reactor cases will be simulated again and the corresponding results analysed to confirm the efficiency of the upgraded calculation schemes and to propose with the guidance of the Senior Expert Group (SEG), harmonized evaluation methodologies for both categories of accidental situations.

2.2 Task 2.1.1 Review of release evaluation methodologies

2.2.1 Task 2.1.1 Objective

This report focuses on task T2.1.1 « Review of release evaluation methodologies » of the WP2- « METHOD ». This task aims at sharing the state of the art of the various release source term evaluation methodologies among partners. This will contribute to identify the major source of uncertainty on the source term evaluation. At the end of the R2CA project, improvements made by the project will be evidenced by comparing the updated source term methodologies developed to the initial ones. The reviewed evaluation methodologies used by different organisations for different reactor types were:

- IRSN : PWR type power plant
- Tractebel and Bel V : PWR type power plant
- VTT : EPR type power plant
- SSTC NRS : VVER type power plant
- ARB : VVER type power plant
- LEI : BWR type power plant

The participants filled out a template where detailing (see [Appendix A](#)):

- The isotopic inventory (i.e. the fuel initial state)
- The source term methodology in LOCA conditions
- The source term methodology in SGTR conditions

2.2.2 Context of each contribution

Each participant shares the available materials used in his the release methodology. As a result, the various methodologies originate from a very different context. Some of them are linked to rather old studies, some are very simple and conservative, some are much more complex, and some partners never performed such an evaluation. In order to place each contribution in its own context, their backgrounds are presented below.

IRSN: most of the hypotheses presented in the report are used in the MER software, developed for probabilistic risk assessment. MER is a ST assessment software for French PWR plants, with similar goal to ASTEC. ASTEC applies state-of-the-art models and simulates a wide range of phenomena (core degradation, FP release from damaged core, thermal hydraulics in the RCS and the containment, detailed iodine chemistry, etc.) and is used to perform computations which require a deeper understanding of the processes occurring during an accident. By contrast, MER uses simplified models in comparison with ASTEC, and is used to perform rapid (order of seconds) Source Term computations with a high degree of confidence for probabilistic risk assessment studies. MER models are validated by ASTEC calculations, which in return are validated by experimental programs conducted in France and abroad. Particularly the iodine chemistry models in MER are up-to-date to the models used in ASTEC version 2.1.1.4. MER can be used in a point-wise or a probabilistic mode, which is why some hypotheses present minimum and maximum values, with a distribution law determining the value of a parameter. It has to be mentioned that ASTEC, and particularly MER have been mainly developed for severe accident scenarios. As a result, until present few calculations have been performed on DBA scenarios using this tool.

Tractebel and Bel V: The only LOCA studies performed by Tractebel are the initial studies performed in the framework of the licensing of the PWR reactors in Doel and Tihange. The methodology used for these studies is based on the content of the US NRC Regulatory Guideline 1.4. Initially a Fortran-code was

created to perform the necessary calculations but at the same time these codes were converted to an easier usable excel-calculation. Since these initial studies, no update has taken place.

Meanwhile, the source term methodology used for SGTR conditions is used within the PWR licensing framework for design basis accidents. This methodology uses the RELAP5 thermal-hydraulics system code to simulate a steam generator tube rupture (SGTR) accident. The developed release model takes into account the main thermal-hydraulic and chemical mechanisms which are: partitioning, flashing, atomization, scrubbing, moisture carry-over and dry-out. To take into account all these mechanisms, a tracking of the activity of noble gases and iodine isotopes across the primary and secondary systems is required (taking into account spiking, dilution, decay, etc.) and implemented within the RELAP5 model.

SSTC NRS: For evaluation of the radioactive release under design basis accidents conservative approaches are used. Two representative accidents (LOCA and primary to secondary leaks) are taken into account in DBA analyses where only the most conservative scenarios with respect to radioactive releases into environment are considered (i.e. double ended MCP rupture (LB LOCA) and steam generator collector rupture). In addition, the following conservative assumptions are used for radioactive release evaluation:

- LB LOCA:
 - o 100% of primary coolant activity released into containment;
 - o Gap activity release into containment;
 - o Design leakage rate (0.3% Vcon./day);
 - o One spray channel operation.
- Primary to secondary leaks:
 - o 100% of primary coolant activity released into environment;
 - o Iodine spike effect is assumed.

For BDBA analyses the radioactive release is evaluated with the MELCOR computer code and with realistic approaches. MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants.

ARB: The methodology used to assess the radiological consequences is determined on the basis of national regulatory document NP 306.2.173-2011 "Requirements for determining the size and boundaries of the observation zone of nuclear power plants". This regulatory document defines the sequence for calculating the meteorological dispersion factor and plume depletion coefficients for instant and long-term releases. In this case, some variation of the parameters is possible, which determines the complexity of the calculation and the degree of conservatism, such as:

1. Characteristics of the release source - point (conservative) or volumetric (realistic).
2. Determination of the effective elevation of release - from the ground level (conservative), from the height of the building or ventilation chimney (realistic) and from the geometric height of the building with account for dynamic elevation of release
3. Determination of the source term taking into account the interfacial transition coefficients, the properties of the containment and mitigation of release within the containment. In this case, the transition coefficient of radionuclides from liquid to vapor and the deposition of the aerosol fraction of radionuclides in the containment may or may not be taken into account.
4. Accounting for the release duration - instant (conservative) or continuous along the progression of the accident (realistic).

To determine the atmospheric dispersion, the Pasquill-Gifford method with some modifications of the coefficients is used.

LEI: Isotopic inventory of BWR fuel is calculated using SCALE code according to the assumptions presented in the report. LOCA accident in the case of DEC-A conditions of the BWR type reactor will be performed by ASTEC code.

All hypothesis presented in the report are based on the initial calculation results of BWR model calculated using ASTEC code, experience gained from the other calculations and engineering judgment. Some of the hypotheses are imposed by the limitations of the developed BWR model, model discretization and modelling choices. The fission product release, their distribution between liquid and gas phase and their speciation will be calculated using ASTEC code, SOPHAEROS module.

VTT: No ST analysis has been performed for the EPR, but the Finnish Radiation and Nuclear Safety Authority (STUK) provides guidelines regarding the ST evaluation. For instance the maximum number of failed fuel rods is set to 10%, the primary coolant activity is set to the maximum value in the Technical Specifications of the plant. All the hypotheses regarding the FP distribution between liquid and gas phase, as well as their speciation shall be justified. When such analysis will be performed, it will be done with the MELCOR code.

3. Source term evaluation methodology in LOCA conditions

In this chapter the aspects of source term methodology in LOCA conditions are discussed. First of all the hypotheses regarding the isotopic inventory, core discretization, the burn-up and the amount of ruptured fuel rods are discussed. Then the FP volatility, their release from the fuel and into the containment and their behaviour in the containment are presented. The effects of the CSS on the FP settling is also discussed. Finally, the release outside the containment (gaseous and liquid) is presented.

Previous work concerning the harmonization of ST methodology practices in LOCA conditions has been presented in the "Determination of the in-containment source term for a Large-Break Loss of Coolant Accident" report ([Ref. 1](#)). The project's goal was to harmonize ST assessment in the containment for PWR plants among TSOs in Belgium, France, Spain, Germany and UK. Other European partners provided feedback on their ST evaluation methodology (for PWR and VVER plants). The report provides proposals regarding the following aspects: accidental sequence definition, list of nuclides, failed fuel fraction, release from failed fuel, retention in the RCS and the initial release of FP to the containment atmosphere and FP behaviour in the containment. This reference is included in the comparison of the present report.

3.1 Isotopic inventory

Table 1 presents a summary of the different approaches used to calculate the isotopic inventory. The majority of participants use a deterministic method and similar codes (SCALE or ORIGEN). For ARB, the isotopic inventory is provided by the fuel supplier, and additional calculations are performed with the aforementioned codes. IRSN uses the internally developed VESTA code, which is a Monte-Carlo code. Concerning the core discretization, IRSN, SSTC NRS and ARB average several different fuel types based on their irradiation history. IRSN and SSTC NRS consider a realistic burn-up distribution for each fuel assembly type, whereas ARB applies a maximum burn-up for each fuel assembly type. For the BWR plant, LEI uses an average fuel bundle with an average burn-up. Concerning Tractebel & Bel V, the LOCA

analysis only considers the most volatile FP, namely iodine and noble gases where their inventories are calculated with ORIGEN-2 code.

Regarding the hypothesis for the number of ruptured fuel rods, IRSN is the least conservative (33% ruptured fuel rods), followed by LEI (55.5%), whereas all the other partners have chosen the most conservative values (100% ruptured fuel rods).

	PWR	VVER	BWR
Code and method used	<u>IRSN</u> : VESTA code (IRSN), Monte-Carlo <u>Tractebel & Bel V</u> : Deterministic, ORIGEN-2 code	<u>SSTC NRS</u> : Deterministic, based on fuel supplier data, additional calculations with specific codes (SCALE, MCNP, etc.) <u>ARB</u> : Supplied by the fuel supplier + additional calculations with ORIGEN and SCALE codes	SCALE code
Discretization	<u>IRSN</u> : Average of 8 different fuel types (based on irradiation and core management) <u>Tractebel & Bel V</u> : Only volatile FP inventory is used in the calculation (ORIGEN-2 for LOCA)	<u>SSTC NRS</u> : Average of several different types of similar fuel assembly <u>ARB</u> : Average of 4 different types of similar fuel assembly (similar irradiation)	One BWR 10x10 fuel bundle
Burn-up considered	<u>IRSN</u> : Realistic burn-up distribution for each assembly type at end of cycle <u>Tractebel & Bel V</u> : 650 days full operation at 3135 MW _{th} (end of cycle);	<u>SSTC NRS</u> : Realistic BU distribution for each assembly type at end of cycle <u>ARB</u> : Maximum BU for each assembly type	Average burn-up
Conditions	<u>IRSN</u> : Power at end of cycle	<u>SSTC NRS</u> : Steady state at full nominal power <u>ARB</u> : Full power at end of cycle	Steady state at full nominal power
% of ruptured fuel rods	<u>IRSN</u> : 33% <u>Tractebel & Bel V</u> : 100%	<u>SSTC NRS</u> : 100% <u>ARB</u> : 100%	55,5%

Table 1. Summary of methodologies for calculating the isotopic inventory and hypothesis on failed fuel fractions

The EUR 19841 EN ([Ref. 1](#)) report proposes in the chapter regarding the sequence definition that the parameters involving the burn-up and the fuel cycle characteristics should be chosen in order to maximise the radiological consequences. Concerning the fraction of failed fuel rods, the proposal was to apply a 33% failed fuel rods for reactor designs with safety injection in the cold leg and 10% failed fuel rods for reactors with injection in both cold and hot legs.

On the other hand, the US NRC Regulatory Guides 1.183 ([Ref. 5](#)) and 1.195 ([Ref. 6](#)) recommend that the FP in the reactor core available for release should be based on maximum full-power operation of the core. All the parameters should be set in order to maximize the FP inventory. The irradiation period should be long enough to allow short-lived FP to reach equilibrium or maximum values. All fuel assemblies in the core are considered to be affected and the core average inventory should be used in the case of a LOCA.

3.2 Elements volatility

Table 2 presents the list of considered isotopes and their classification into four groups based on their volatility: noble gases, volatile elements, semi-volatile elements and low-volatile elements. One major difference is that every project partner is considering a different number of FPs in their analysis.

Regarding the classification of FPs, the differences are minor, particularly for the volatile FPs, where SSTC NRS considers tellurium (Te) to be semi-volatile, while it is classified as volatile by the other partners. Other differences concern lanthanum (La), ruthenium (Ru) and cerium (Ce), which are classified either semi-volatile or low-volatile.

	IRSN	Tractebel/Bel V	SSTC	ARB	LEI
Xe	noble gas	noble gas	noble gas	noble gas	noble gas
Kr	noble gas	noble gas	noble gas	noble gas	noble gas
Cs	Volatile		volatile	volatile	volatile
Rb	Volatile				volatile
Te	Volatile		semi-volatile		volatile
I	Volatile	volatile	volatile	volatile	volatile
Br	Volatile				volatile
Ag					volatile
Sr	semi-volatile		semi-volatile	semi-volatile	semi-volatile
Ba	semi-volatile		semi-volatile		semi-volatile
La	semi-volatile		low volatile	low volatile	semi-volatile
Eu	semi-volatile				semi-volatile
Ru	low volatile		low volatile	semi-volatile	low volatile
Ce				low volatile	semi-volatile
Pu			low volatile		low volatile

Table 2. List of considered FP for LOCA analysis and their volatility

The [Ref. 1](#) report proposes that all FP should be considered and if some FPs are neglected, it should be justified correspondingly. It also stresses the importance of considering the actinides, mostly for general public acceptance. They might be ignored, but it should be thoroughly justified.

3.3 Fuel release

The FP release rates from the fuel to the reactor coolant system (RCS) are summarized in Table 3. In most cases, release to the coolant is either not considered (IRSN) or it is not distinguished from the release to steam (Tractebel & Bel V and ARB). The values are with respect to either the initial inventory or the gap inventory.

As such, the FP release rates used by IRSN will depend on the nature of the FP, as well as the burn-up. Tractebel & Bel V use a conservative approach and consider an integral release of iodine and noble gases (xenon and krypton) from the fuel. SSTC NRS and ARB assume a 100% release of FP located in the gap between the fuel pellet and the fuel cladding. For SSTC NRS, 100% gap release corresponds to 0.4 % of initial inventory for noble gases and 0.4 – 1.8 % for volatile FPs. ARB also distinguishes between short-lived (SL) and long-lived (LL) FP. The approach used by LEI consists in calculating the fission product release and distribution between liquid and gas phase and their speciation using ASTEC code, SOPHAEROS module

	PWR		VVER		BWR	
	Fuel to coolant	Fuel to steam	Fuel to coolant	Fuel to steam	Fuel to coolant	Fuel to steam
Noble gas	IRSN : - Tractebel & Bel V : 0%	IRSN : 2- 25% Tractebel & Bel V : 100%	SSTC NRS : 100% gap release	SSTC NRS : 100% TCC	100%, will be calculated by SOPHAEROS	100%, will be calculated by SOPHAEROS
			ARB : 100% TCC with spike effect	ARB : 100% gap release		
Volatile FP	IRSN : - Tractebel & Bel V : 50%	IRSN : 2-8% Tractebel & Bel V : 50%	SSTC NRS : 100% gap release	SSTC NRS : 100% TCC	SL - 0.5% LL – 1%, will be calculated by SOPHAEROS	SL - 0.5% LL – 1%, will be calculated by SOPHAEROS
			ARB : 100% TCC with spike effect	ARB : 100% gap release		
Semi- volatile FP	IRSN : - Tractebel & Bel V : -	IRSN : 0.01% Tractebel & Bel V : -	SSTC NRS : 100% gap release	SSTC NRS : 100% TCC	SL - 0.5% LL – 1%, will be calculated by SOPHAEROS	SL - 0.5% LL – 1%, will be calculated by SOPHAEROS
			ARB : 100% TCC with spike effect	ARB : 100% gap release		
Low volatile FP	IRSN : - Tractebel & Bel V : -	IRSN : 0.01% Tractebel & Bel V : -	SSTC NRS : 100% gap release	SSTC NRS : 100% TCC	SL - 0.5% LL – 1%, will be calculated by SOPHAEROS	SL - 0.5% LL – 1%, will be calculated by SOPHAEROS
			ARB : 100% TCC with spike effect	ARB : 100% gap release		

Table 3. FP release rate from the fuel into the RCS

The reference report does not provide any recommendations regarding the release of FP from the fuel, since the project partners had not agreed upon a common position. France, Germany and UK used a best estimate approach, which was recommended in a previous report (Ref. 2), whereas Spain and Belgium maintained a very conservative approach stipulated in the Regulatory Guide 1.4 (Ref. 3).

3.4 Release in the containment

Table 4 presents the retained hypotheses for the FP release in the containment. SSTC NRS, ARB and LEI use a conservative approach and assume that 100% of the FP (both noble gases and the aerosol species) are released to the gas phase of the containment. IRSN considers a 100% release of noble gases to the gas phase of the containment and aerosols are distributed 10% and 90% to the gas phase and liquid phase respectively. Tractebel & Bel V also consider a 100% transfer of noble gases to the gas phase of the containment. However, for iodine it is assumed that 50% goes to the liquid phase and 50% to the gas phase, of which half is bound to the containment walls, which means that 25% of the initial iodine inventory is in the gas phase of the containment.

No retention in the RCS is being considered either in this project, or in the previous one (Ref. 1); all of the FP released from the damaged fuel are assumed to be released in the containment.

		PWR	VVER	BWR
Noble gas	Gas phase	IRSN : 100% Tractebel & Bel V : 100%	SSTC NRS : 100% gap release ARB : 100%	100%
	Liquid phase	IRSN : 0% Tractebel & Bel V : 0%	SSTC NRS : 100% TCC ARB : 0%	0%
Aerosols	Gas phase	IRSN : 10% Tractebel & Bel V : 25% (+ 25% on walls)	SSTC NRS : 100% gap release ARB : 100%	100%
	Liquid phase	IRSN : 90% Tractebel & Bel V : 50%	SSTC NRS : 100% TCC ARB : 0%	0%

Table 4. FP release from the RCS to the containment

The chemical form of iodine released in the containment is presented in table 5. The IRSN assumption is that a small fraction (5%) of iodine is released as gaseous molecular iodine (I_2) and the rest (95%) is released metallic iodine aerosol (Csl). SSTC NRS and ARB are using an instantaneous modelling of the chemistry in the containment, which is why they assume that the iodine form at the breach equals to the average iodine form in the containment. That is the reason why the organic iodine species (RI) is additionally considered in their hypothesis, albeit in different ratios (1% for SSTC NRS and 4% for ARB). Furthermore, they consider a quite similar molecular iodine (I_2) fraction to other participants (respectively 4% and 5%). For the BWR case, the speciation will be calculated by the SOPHAEROS module of the ASTEC code. Tractebel & Bel V methodology does not take into account the iodine speciation for the analysis.

	PWR	VVER	BWR
Molecular iodine (I_2)	IRSN : 5% Tractebel & Bel V : --	SSTC NRS : 4% ARB : 5%	As calculated by SOPHAEROS
Particular iodine (Csl)	IRSN : 95% Tractebel & Bel V : --	SSTC NRS : 95% ARB : 91%	As calculated by SOPHAEROS
Organic iodine (RI)	IRSN : -- Tractebel & Bel V : --	SSTC NRS : 1% ARB : 4%	--

Table 5. Iodine speciation at the breach

3.4.1 Summary

If it were to compare the weight of each hypothesis on the FP release in the containment, Figures 1 and 2 show that while the assumption on the number of ruptured fuel rods has an impact on the quantity of FPs present in the gaseous phase of the containment, other hypotheses, such as the FP release rate from the fuel, are equally important.

Concerning the noble gases, Tractebel & Bel V show the highest amount of released Xe and Kr, which is due to the conservative hypothesis of a 100% release rate from the fuel. The next most important release is shown by IRSN with the 25% release rate, which is the highest value in the 2-25% release rate hypothesis used in the MER code (see Table 3), despite having only 33% of ruptured fuel rods.

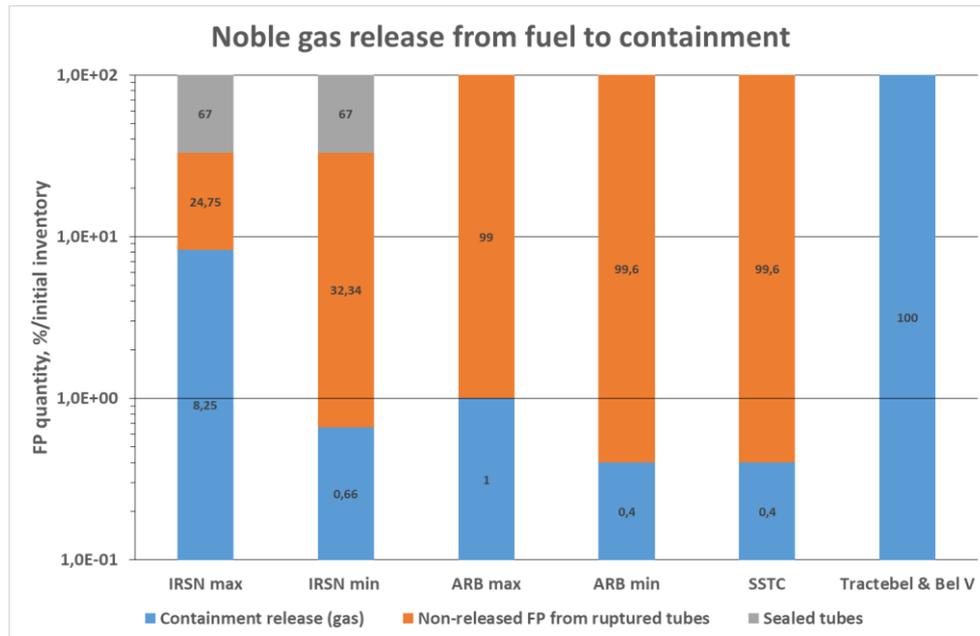


Figure 1. Relative amount of Xe and Kr in the sealed (grey) and ruptured (orange) fuel rods and released to the gas phase of the containment (blue)

The same observations apply to the release of volatile FP (Figure 2). Tractebel & Bel V show the highest amount of iodine in the gas phase of the containment, which is mainly due to fuel release rate and number of failed fuel rods hypotheses. Additionally in the case of volatile FP the partitioning between the liquid and gas phases plays a significant role. As such, even though the IRSN hypothesis assumes a higher fuel release rate than SSTC NRS and ARB (2-8% vs. 0.4-1.8%), the amount of volatile FP present in the gas phase is lower, due to the transfer to the liquid phase.

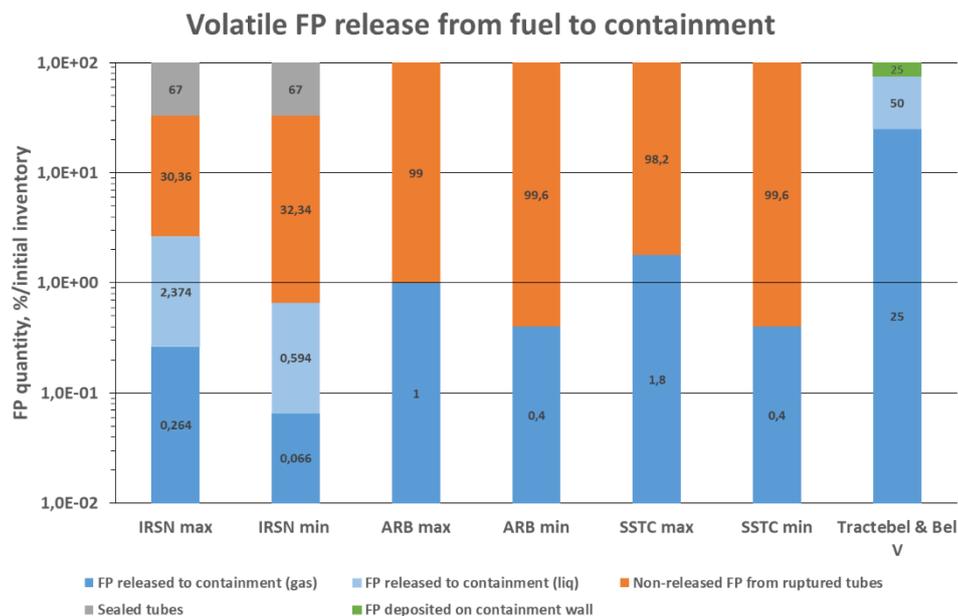


Figure 2. Relative amount of volatile FP in the sealed (grey) and ruptured (orange) fuel rods and released to the gas (blue) and liquid (light blue) phases of the containment

3.5 Fission product behaviour in the containment

3.5.1 Fission product chemistry

The FP behaviour in the containment concerns the production, the consumption and the mass transfer between the gas and liquid phases of the containment of iodine containing species such as molecular iodine (I_2), metallic iodide aerosols (Csl), organic iodine (RI) and iodine oxides (IOx) as well as noble gases and other aerosols. Two approaches are considered: instantaneous chemistry modelling and time-dependent modelling.

For the VVER reactor types, SSTC NRS and ARB consider no production or consumption of iodine species via chemical reactions. The iodine speciation in the containment stays identical to the speciation at the breach. For the VVER case, it is assumed conservatively that all the released FP at the breach remain in the gas phase of the containment and no liquid-gas transfer occurs. For the PWR reactor type, Tractebel & Bel V also do not consider any production or consumption of iodine, except the plate out phenomenon, where 25% of the iodine released in the gas phase of the containment is trapped by the walls. Additionally, Tractebel & Bel V considers no liquid-gas mass transfer.

IRSN (PWR) and LEI (BWR) are considering instead time-dependent chemistry modelling. IRSN applies a reduced set of chemical reactions used by the SOPHAEROS module of the ASTEC code to determine the iodine chemistry in the gas phase (for I_2 , RI and IOx) and liquid phase (for I_2 and I⁻). LEI use SOPHAEROS as well to establish the iodine chemistry in the gas and liquid phases of the containment.

3.5.2 Containment Spray System impact

The impact of the containment spray system (CSS) on the amount of FP in the gas phase is illustrated in Figures 3 & 4. For molecular iodine (I_2), SSTC NRS apply the most conservative approach and consider no CSS operation (Figure 3). ARB employs a similar approach, however, if the results are too conservative, the operation of the CSS might be taken into account. For the BWR reactor, LEI assumes that the CSS is not operational. In fact, the selected accident scenario does not meet the conditions required for the activation of the CSS. For the PWR reactor case, IRSN and Tractebel & Bel V consider the operation of the CSS, with an exponential law of decreasing I_2 concentration, which has the following general expression:

$$A(t) = A_0 \times e^{-\lambda t}$$

The Tractebel & Bel V hypothesis assumes two different λ values, one for the injection phase (1.05 h^{-1}) and one for the recirculation phase (0.31 h^{-1}). When the I_2 concentration decreases by a factor of 100, the CSS is no longer taken into account. The IRSN hypothesis considers a greater value for the λ parameter (2.7 h^{-1}), which takes into account that the water is alkaline, as well as quicker time for the CSS activation, hence the I_2 concentration decreases more rapidly. IRSN also applies a lower threshold until the CSS is not taken into account (factor 10 000).

The same observations apply with regard to the aerosols settling. SSTC NRS, ARB and LEI apply the most conservative hypothesis and don't consider the CSS operation, except ARB which applies a settling coefficient in some cases. Tractebel & Bel V modelling does not take into account the aerosols (all iodine is present in gas form). IRSN applies an exponential law of decreasing aerosols concentration, with $\lambda = 1.3 \text{ h}^{-1}$ for $t < 0.6 \text{ h}$ and $\lambda = 0.5 \text{ h}^{-1}$ for $t > 0.6 \text{ h}$, which is lower compared to the I_2 settling. As a result, the aerosol settling is less efficient. This is due to the fact that the CSS system is connected to a tank containing sodium hydroxide, which is efficient in trapping molecular iodine in the liquid phase.

The CSS operation is considered inefficient towards the settling of noble gases for all the reactor cases.

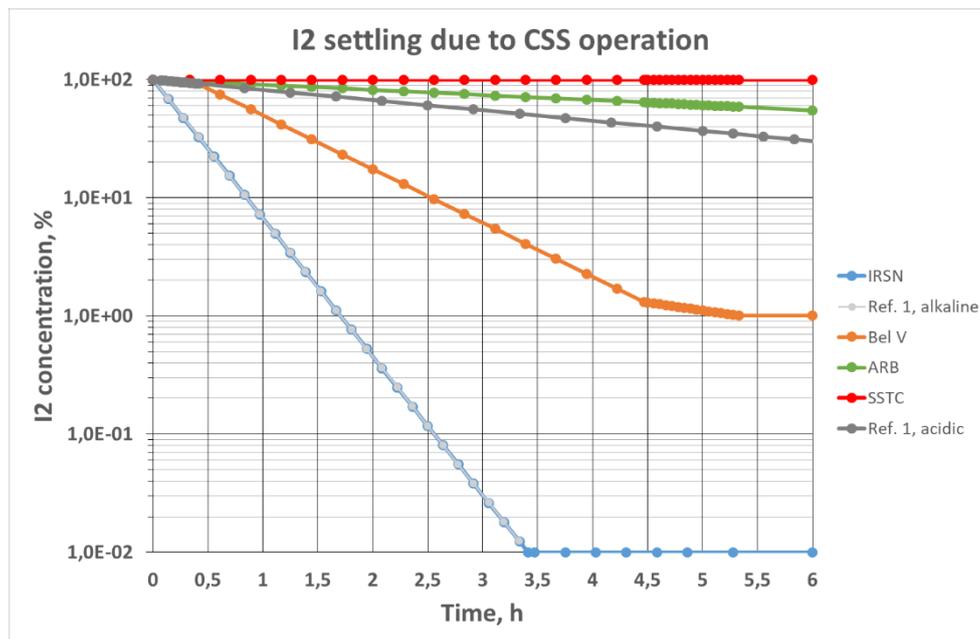


Figure 3. Molecular iodine (I_2) relative concentration in the gas phase of the containment in the event of Containment Spray System operation

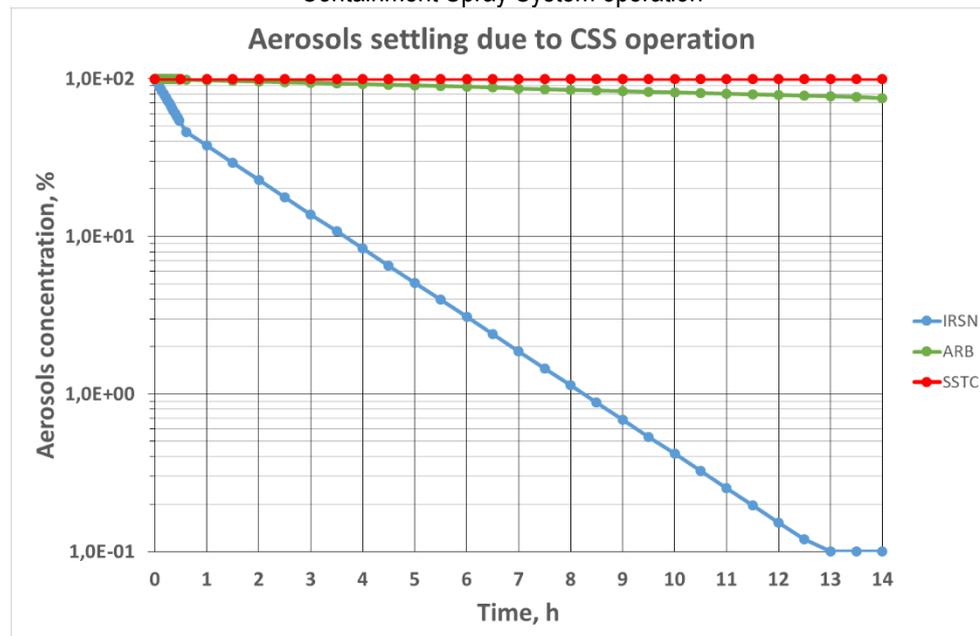


Figure 4. Aerosols relative concentration in the gas phase of the containment in the event of Containment Spray System operation

The EUR 19841 EN ([Ref. 1](#)) proposes that in the case of reactors with cold leg safety injection with CSS, a specific iodine code should be used, which would take into account a realistic iodine release kinetic model during the dry and wet phase, as well as iodine removal by spray and adsorption on the containment wall. In case such a code is unavailable, they suggest that in addition to the amount of I_2 released during the dry phase, 60% of the iodine released during the wet phase should be converted to I_2 and released directly to the gas phase of the containment. Regarding the I_2 removal by spray, they propose the following removal constants based on the spray water pH: $\lambda = 0.2 \text{ h}^{-1}$ for acidic spray and $\lambda = 2.7 \text{ h}^{-1}$ for alkaline water spray.

For organic iodine, it is proposed that 0.2% of the iodine released during the dry phase should be converted into organic iodine, instantaneously present in the gas phase of the containment. No removal is to be applied to this species other than the radioactive decay.

Concerning the aerosols, the report advises using a detailed mechanistic aerosol behaviour model, which would take into account different removal mechanisms as a function of aerosol sizes and considering an aerosol size distribution. If such a model is not available, a total removal coefficient should be applied, as proposed in an earlier report ([Ref. 2](#)).

3.5.3 Decay and filiation

The FP decay is taken into account in the PWR and VVER reactor cases and ignored for the BWR (Table 6). The FP filiation is only taken into account for the PWR case, and Tractebel & Bel V only consider the iodine-xenon filiation but not the filiation that produces iodine (ex. ^{131}Te decay to ^{131}I).

	PWR	VVER	BWR
FP decay	IRSN : yes Tractebel & Bel V : yes	SSTC NRS : yes ARB : yes	no
FP filiation	IRSN: yes Tractebel & Bel V: yes	SSTC NRS : no ARB : no	no

Table 6. Hypotheses regarding FP decay and filiation

3.6 Containment releases

3.6.1 Unfiltered releases

The hypotheses regarding the unfiltered gas releases from the containment building to the environment are presented in Table 7 and illustrated in Figure 5. For all participants, the same leak rate is applied to all the species (noble gases, molecular iodine, organic iodine and aerosols).

For the French single containment reactors (900 MWe) all the releases are assumed to go into the environment, bypassing the auxiliary buildings. For the double containment units (1300 MWe), 1.35% vol/day is collected by the annular space and 0.15% vol/day are leaks directly to the environment for the first 4 days of the accident, then the leak rate is assumed to be 4.5% vol/day (4.35% vol/day collected by the annular space and 0.15% vol/day leaked through the penetrations). For the Belgian units, since all the reactors are equipped with a double containment, in general primary containment leaks are collected in an annular space which is kept under negative pressure w.r.t. the environment. The primary containment leaks are captured by the ventilation system of the annular space and discharged via the stack. For the secondary containment no leaks are considered (since the annular space is at a lower pressure than the environment).

However some leaks to the outside are defined since some piping is installed through the penetrations of both the primary and secondary containment which causes that these leaks will bypass the annular space. For these bypass leaks, conservative flowrate values are chosen which are taken into account in the Tech. Spec.

For the VVER it is assumed that 100% of the leaks go into the environment.

IRSN single containment	Nikuradze law : $\frac{F_{max}}{F_1} = \left(\frac{P_{max}^2 - P_0^2}{P_1^2 - P_0^2} \right)^{1/2} \times \frac{P_1}{P_{max}}$, with F_{max} the maximum leak rate (0.3% vol/day) at P_{max} (4.85 bar abs), F_1 the leak rate at P_1 and P_0 atmospheric pressure
IRSN double containment	Laminar flow law : $\frac{F_{max}}{F_1} = \frac{P_{max}^2 - P_0^2}{P_1^2 - P_0^2} \times \frac{P_1}{P_{max}}$, with F_{max} the maximum leak rate (1.5% vol/day) at P_{max} (5 bar abs), F_1 the leak rate at P_1 and P_0 atmospheric pressure; 1.5% vol/day for $t < 96h$, then 4.5% vol/day
Tractebel & Bel V	0,25% vol/d for $t < 24h$ and 0,125% vol/d for $t > 24h$ for Doel NPP 0,5% vol/d for $t < 24h$ and 0,25% vol/d for $t > 24h$ for Tihange NPP
SSTC NRS	$Q_i \times k_1$, with Q_i – activity of isotope « i » in containment and $k_1 = \int_0^{t_1} (1 - k_2^1 \cdot t) \cdot \frac{k_2^1}{24 \cdot 3600} \cdot e^{-\lambda \cdot t} dt + \int_{t_1}^{t_{14}} (1 - 0.003 - k_2^{14} \cdot t) \cdot \frac{k_2^{14}}{24 \cdot 3600} \cdot e^{-\lambda \cdot t} dt$ $k_2^1 = 0,003$ – containment leakage during first day $k_2^{14} = 0,0015$ - containment leakage during 2-14 days
ARB	0.3%/day ($t < 24 h$) 0.15%/day ($24 h < t < 14 d$) The estimates are made with account for decrease of FP concentration due to decay. In some cases, FP deposition in the containment also can be considered, including impact of the CSS
LEI	0,5% vol/day

Table 7. Containment release rates - unfiltered gas releases

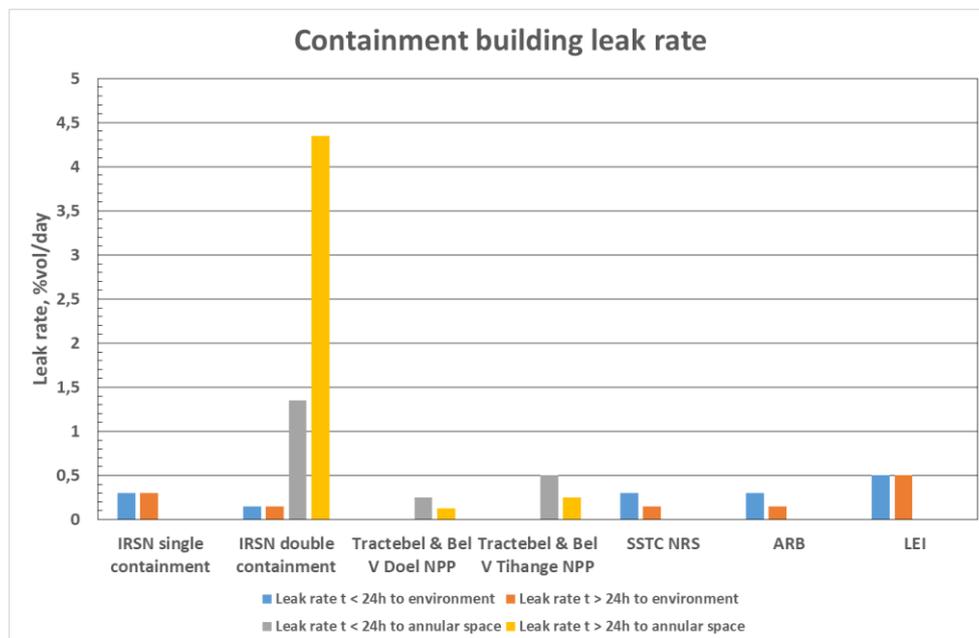


Figure 5. Containment building leak rates for unfiltered gas releases

The impact of the leak rate on the release of noble gases into the environment, at one and seven days, can be observed in Figures 6 and 7. Overall, the initial hypothesis on the release ratio will have the greatest impact on the amount of released FP (i.e. IRSN max column). However, for similar fuel release

hypotheses, the leak rate will have a larger impact. For instance, after 7 days, ARB and SSTC NRS releases have increased by a factor of 4 (compared to the first 24h), whereas IRSN releases increased by a factor of 7.

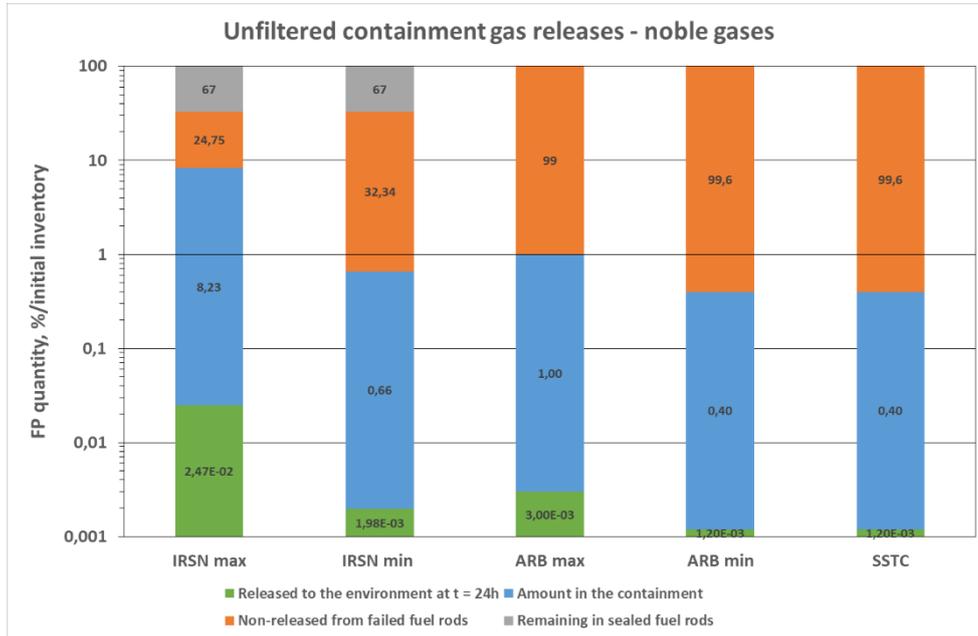


Figure 6. Unfiltered noble gases releases from the containment at t = 24 h (IRSN single containment units)

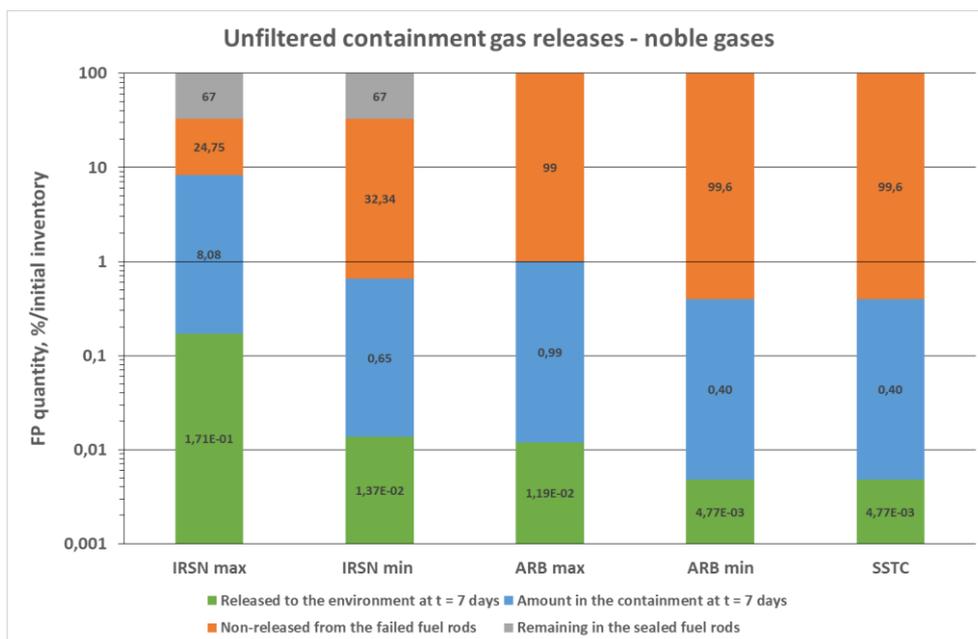


Figure 7. Unfiltered noble gases releases from the containment at t = 7 days (IRSN single containment units)

The unfiltered iodine (gaseous and particulate) releases are presented in Figure 8 (the maximum release from fuel hypothesis is presented)¹. Unlike the noble gases, ARB and SSTC NRS show a greater release to the environment of both gaseous (I₂, RI) and particulate iodine. The reason is that the iodine settling

¹ A penalizing assumption was made regarding the operation of the CSS: before reaching the concentration threshold values (i.e. C/C₀ = 10⁻⁴ for I₂) the iodine concentration is assumed to be maximal; after that point in time the concentration is minimal.

due to CSS operation is not taken into account, whereas in IRSN hypothesis, molecular iodine and aerosols are efficiently settled in a few hours after the start of the accident.

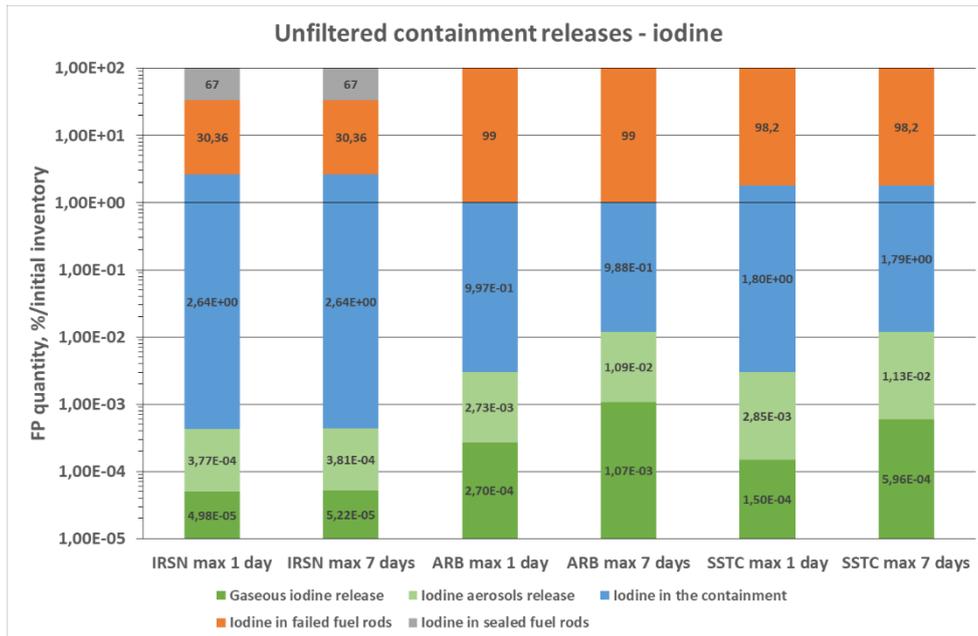


Figure 8. Unfiltered gaseous (dark green) and particulate (light green) iodine release from the containment to the environment after 24h and 7 days (IRSN single containment units)

The unfiltered liquid releases are considered only for the PWR reactor case. Tractebel & Bel V consider liquid leaks of the containment only through certain penetrations. Similarly to the gaseous leaks, the value used for the liquid bypass leaks is based on the value in the Tech. Spec. IRSN does not consider direct liquid containment leaks, instead it considers liquid leaks to the auxiliary buildings. They are distributed as follows:

- Leaks to the auxiliary safeguard building : 200L/min since the start of the recirculation phase, which can be isolated in 30 min ; after that, 7L/min ; the leaks are supposed to be caused by sealing problems in the Safety Injection System and the CSS ;

Leaks to the Reactor Cavity and Spent Fuel Pit Cooling and Treatment System: two types of leaks are considered. The first type concerns liquid leaks in the gas phase of the tank, and the second one leaks into the liquid phase of the tank. The values for this parameter are presented in Table 8.

	Leak to the gas phase	Leak to the liquid phase
PWR 900	28L/h (1h < t < 24h) 27L/h (t > 24h)	72L/h (1h < t < 24h) 12L/h (t > 24h)
PWR 1300	36L/h (1h < t < 24h) 27L/h (t > 24h)	52L/h (1h < t < 24h) 39L/h (t > 24h)

Table 8. Liquid leaks from the containment sump to the Reactor Cavity and Spent Fuel Pit Cooling and Treatment System

Concerning the liquid leaks outside the containment, for iodine the revolatilisation from liquid phase has to be considered. The liquid to gas mass transfer rate of iodine containing species is 1% for I₂, RI and IO_x in the hypothesis applied by Tractebel & Bel V whereas in the IRSN approach, the mass transfer rate is calculated by SOPHAEROS for I₂ only.

3.6.2 Filtered releases

Along with unfiltered releases to the environment, there are also leaks that are collected by the ventilation systems and filtered before being released to the environment. Table 9 presents the values for the decontamination factor for the iodine species, the noble gases and the aerosols.

	Noble gas	Molecular iodine (I ₂)	Organic iodine (RI)	Aerosols
PWR	IRSN : 1 Tractebel & Bel V : 1	IRSN : 1000 Tractebel & Bel V : 10	IRSN : 100 Tractebel & Bel V : 10	IRSN : 1000 Tractebel & Bel V : NA
VVER	SSTC NRS : according to filtered venting supplier data ARB* : 1	SSTC NRS : according to filtered venting supplier data ARB : 100	SSTC NRS : according to filtered venting supplier data ARB : 1	SSTC NRS : according to filtered venting supplier data ARB : 1000
BWR	1	1	1	1

Table 9. Decontamination factor for different iodine containing species of the filtering systems

For the VVER reactor case, filter accounting depends on the number of filters and their characteristics. For a standard system of two filters, the values are shown in the table.

Figure 9 presents the global release of iodine in the environment, with respect to the initial inventory, and takes into account the unfiltered releases from the containment building, as well as the filtered releases that originate in the auxiliary safeguard buildings from the liquid leaks from the extended third barrier. Concerning IRSN hypotheses, the decontamination factor applied in the ventilation and filtering systems reduces substantially the amount of released iodine, with an overall global release which is lower compared to ARB and SSTC NRS.

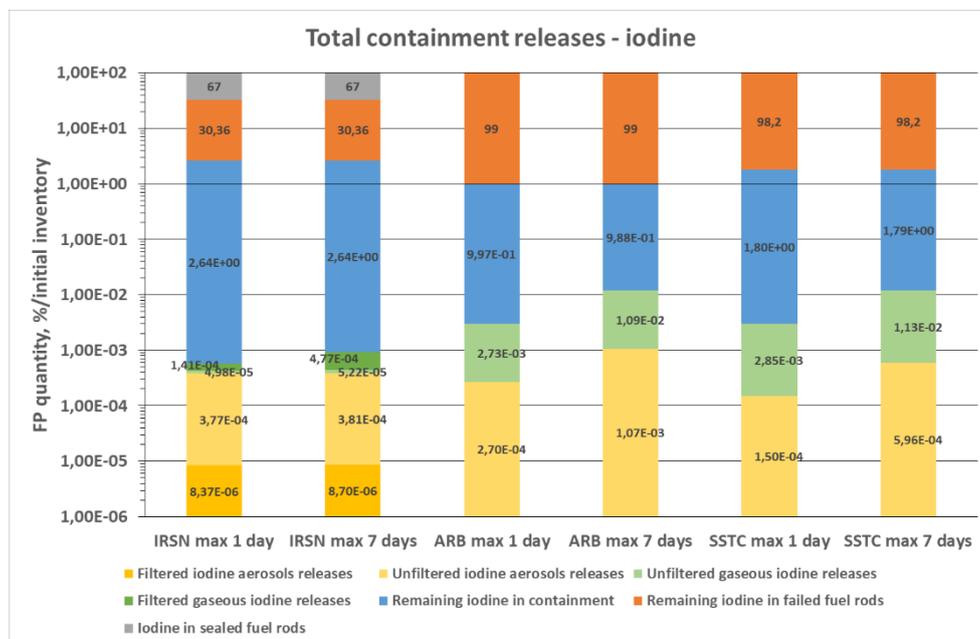


Figure 9. Filtered (dark) and unfiltered (light) gaseous (green) and particulate (yellow) iodine releases to the environment after 24h and 7 days

4. Source term methodology in SGTR conditions

In this chapter are discussed the aspects of source term methodology in SGTR conditions. These include the primary loop activity, the iodine speciation in the primary loop, the FP transfer to the liquid and gas phases of the steam generator (SG) and the subsequent retention in the secondary loop and release into the environment.

A similar task has been performed for the EPR reactor in the context of the Multinational Design Evaluation Programme (Ref. 4). The R2CA project expands on the previous work, by analysing the PWR as well as the VVER reactor types.

4.1 Primary loop activity

The activity of the primary loop coolant is evaluated differently depending on the reactor concept and the organisation. For the VVER reactor, both SSTC NRS and ARB consider the activity of a wide range of FP, while ARB generally assumes more conservative values (see Tables 10 & 11).

For the PWR reactor case, Tractebel & Bel V only consider ^{131}I , and a pre-spiking is postulated prior to the SGTR, such as the primary activity raise to the maximum allowed value as specified by the technical specifications.

IRSN uses a slightly different approach regarding iodine. Instead of considering a specific iodine isotope, the activity of « equivalent ^{131}I » is taken into account, which is the weighted sum of all the iodine isotopes present in the primary coolant and which is calculated using the following equation :

$$A(Eq. I131) = A(I131) + \frac{A(I132)}{30} + \frac{A(I133)}{4} + \frac{A(I134)}{50} + \frac{A(I135)}{10}$$

Then the maximum allowed value by the technical specifications is used for SGTR scenario, which is 150 GBq/t. This approach is used from a radiological consequences point view, however, the value of $A(Eq. ^{131}\text{I})$ decreases in time, due to the short-lived iodine isotopes. For example, the ratio $A(Eq. ^{131}\text{I})/A(^{131}\text{I}) = 29$ at $t = 0\text{h}$ and 4.1 at $t = 24\text{h}$. For the other FP, IRSN uses the penalized NPP operational feedback. Figure 10 illustrates the different values used for ^{131}I primary coolant activity by the project partners.

Primary loop activity	PWR			VVER				
	IRSN		Tractebel & Bel V	SSTC NRS		ARB		
	Mass activity (Bq/kg)	Mass activity (Bq/kg) spike	Mass activity (Bq/kg)	Mass activity (Bq/kg)	Mass activity (Bq/kg) spike	Mass activity (Bq/kg)	Mass activity (Bq/kg) spike	
Kr85	5E+06	5E+06		100% TCC (Typical Coolant Concentrations) + 30% iodine Spike		2E+04	2E+04	
Kr85m	5E+06	5E+06				9E+06	3E+07	
Kr87	5E+06	5E+06				9E+06	8E+07	
Kr88	8E+06	8E+06				2E+07	8E+04	
Xe133	2E+08	2E+08				8E+07	1E+08	
Xe135	3E+07	3E+07				2E+07	4E+07	
Xe135m	4E+06	4E+06				3E+06	2E+07	
Sr90	1E+03	1E+03				4E+01	4E+01	
Ru103						7E+01	7E+01	
Ru106	2E+06	2E+06				4E+00	4E+00	
I131	1E+07	8E+07	Maximum Technical Specifications				4E+06	6E+07
I132	4E+06	2E+07					9E+06	2E+08

I133	2E+07	7E+07			1E+07	1E+08
I134	3E+06	3E+06			6E+06	1E+08
I135	6E+06	2E+07			8E+06	1E+08
Cs134	3E+06	3E+07			4E+05	7E+05
Cs137	4E+06	6E+06			6E+05	1E+06
Ce141					5E+02	5E+02
Ce144					3E+01	3E+01
La140	5E+05	2E+08			5E+03	5E+03

Table 10. Primary loop activity

Nuclide	Coolant activity, Bq/kg	
	Without spike	With iodine spike
Xe-133	4.66E7	6.03E7
Cs-134	2.49E5	3.81E5
Cs-137	3.75E5	5.67E5
I-131	2.23E6	3.82E7

Table 11. Typical coolant concentrations (TCC) for some nuclides in the Ukrainian VVER types reactors

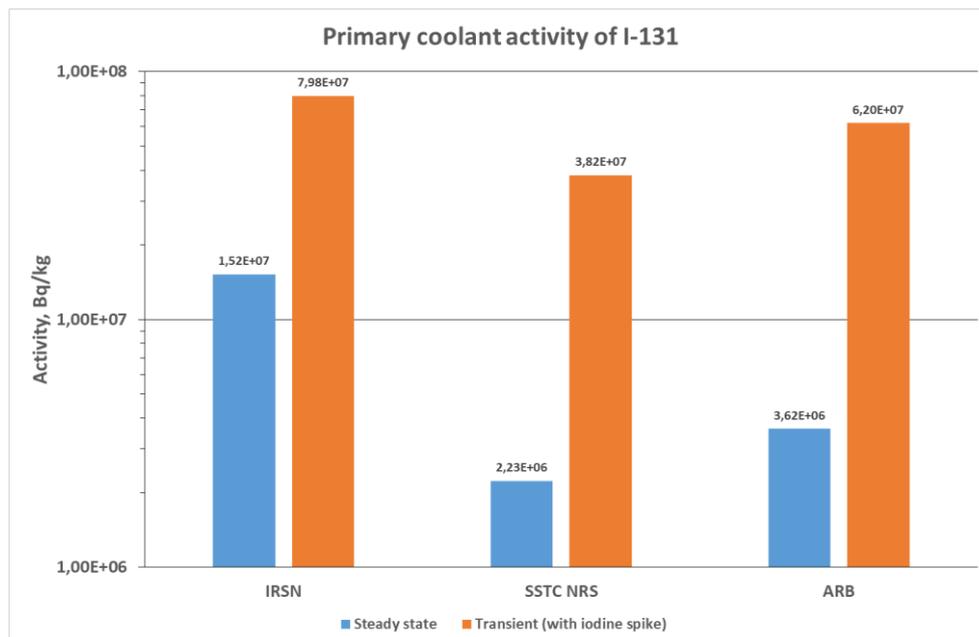


Figure 10. Primary coolant activity of I-131 in steady state (blue) and transient (orange) regimes

4.2 Iodine form in the primary loop

Table 12 presents the assumptions regarding the chemical form of iodine in the primary loop. There are noticeable differences between the two reactor cases. For the PWR, IRSN considers that 0.1% of iodine is present in I_2 form, while the resting 99.9% is in CsI form. In the Tractebel & Bel V modelling, the radiological model transports ^{131}I without discriminating the chemical form carrying the iodine.

The hypotheses for the VVER reactor are the same as for the iodine speciation in the containment during a LOCA with a fraction of iodine being assumed in organic form varying from about 25% (for SSTC) to 50% (for ARB) of the volatile iodine.

	PWR		VVER	
	IRSN	Tractebel & Bel V	SSTC NRS	ARB
Molecular iodine (I ₂)	0.1%	100%	4%	5%
Metallic iodide aerosols (CsI)	99.9%	NA	95%	91%
Organic iodine (RI)	NA	NA	1%	4%

Table 12. Iodine chemical form in the primary loop coolant during SGTR

4.3 Transfer in the Steam Generator

Table 13 presents the transfer of FP to the gas phase of the SG. SSTC NRS has a very conservative approach and assumes that all FP (noble gas, molecular iodine and other forms of FP) are released integrally to the gas phase of the SG. Same assumptions are used by ARB in their “conservative case” for noble gases and molecular iodine. While IRSN and ARB (in “realistic case”) both consider 100% of noble gas to be released to the gas phase, they use a less conservative approach regarding other FP: 100% and 2% of I₂ released to the gas phase for IRSN and ARB respectively and 1% (IRSN) and 0.64-1.38% (ARB) for aerosols.

As mentioned in the previous section, Tractebel & Bel V do not consider any other FP except ¹³¹I. Its distribution to the gas phase is calculated from the contribution of two phenomena: isenthalpic flashing and droplet atomisation. The flashing contribution - in case the primary coolant is superheated with respect to the secondary side (due to the pressure difference), a fraction *x* of the break flow rate will vaporise instantaneously by an isenthalpic process called flashing. It is supposed that the specific activity of the flashed steam (Bq/kg), and the droplets, is equal to the primary specific activity. The *x* fraction is fixed by the isenthalpic process:

$$x = \frac{h_{break} - h_{liqSG}}{h_{vapSG} - h_{liqSG}}$$

where *h_{break}*: break enthalpy, *h_{liqSG}*: enthalpy of the liquid downstream break and *h_{vapSG}*: enthalpy of the steam downstream break. The atomisation fraction *y* = part of the liquid phase (so not flashed, i.e. (1-*x*)) issued from the break that escapes from the affected SG under form of droplets. Not calculated by the TH code, but imposed by following rules: Net Liquid Height (NLH) is first calculated above the break:

- if NLH > 2ft + uncertainty (due to boiling), *y*=1% (covered break) ;
- if NLH ≤ 2ft + uncertainty (due to boiling), *y*=30% (uncovered break) ;

The abovementioned hypotheses for the PWR are valid in the case of a breach situated in the steam phase of the SG and thus the source term is more conservative compared to a breach in the water phase of the SG.

Table 14 presents, on the other hand, the FP transfer to the liquid phase of the SG. All the participants consider that all the activity that was not transferred to the gas phase was transferred to the liquid phase of the SG, thus the numbers represent the result of the following subtraction: 100% - gas phase transfer.

	PWR		VVER	
	IRSN	Tractebel & Bel V	SSTC NRS	ARB
Noble gas	100%	NA	100%	100%
Molecular iodine (I ₂)	100%	Isoenthalpic flashing + droplet atomisation	100%	100% (conservative) 2% (realistic)
Aerosols	1%	NA	100%	0.64-1.38%

Table 13. FP transfer to the gas phase of the Steam Generator for an uncovered break

	PWR		VVER	
	IRSN	Tractebel & Bel V	SSTC NRS	ARB
Noble gas	0%	NA	0%	0%
Molecular iodine (I ₂)	0%	Remaining activity (see Table 13)	0%	0% (conservative) 98% (realistic)
Aerosols	99%	NA	0%	98.62-99.36%

Table 14. FP transfer to the liquid phase of the Steam Generator for an uncovered break

		PWR		VVER	
		IRSN	Tractebel & Bel V	SSTC NRS	ARB
Noble gas	Gas	100%	NA	100%	100%
	Liquid	0%		0%	0%
Molecular iodine (I ₂)	Gas	0.1%	NA	4%	0.1% (realistic) 5% (conservative)
	Liquid	0%		0%	4.9% (realistic) 0% (conservative)
Organic iodine (RI)	Gas	NA	NA	1%	4%
	Liquid			0%	0%
Aerosols	Gas	0.999%	NA	95%	0.58-1.25%
	Liquid			98.90%	0%

Table 15. FP species and phase distribution in the SG with respect to the total activity of the FP in the SG

Figure 11 synthesizes the previous information and presents the partitioning between the gas and liquid phases of the SG of different iodine containing species. With between 10% to 37% of the break flow is directly delivered towards the environment once a path towards the environment is accessible, Tractebel & Bel V have the most important source term, followed by SSTC NRS which use a 100% transfer to the gas phase assumption. Consequently IRSN and ARB show the least important source term, from a relative standpoint. The methodology used by IRSN for calculating the initial inventory (i.e. $A(\text{Eq. }^{131}\text{I})$ vs. $A(^{131}\text{I})$) might influence the result if one looks at the absolute values.

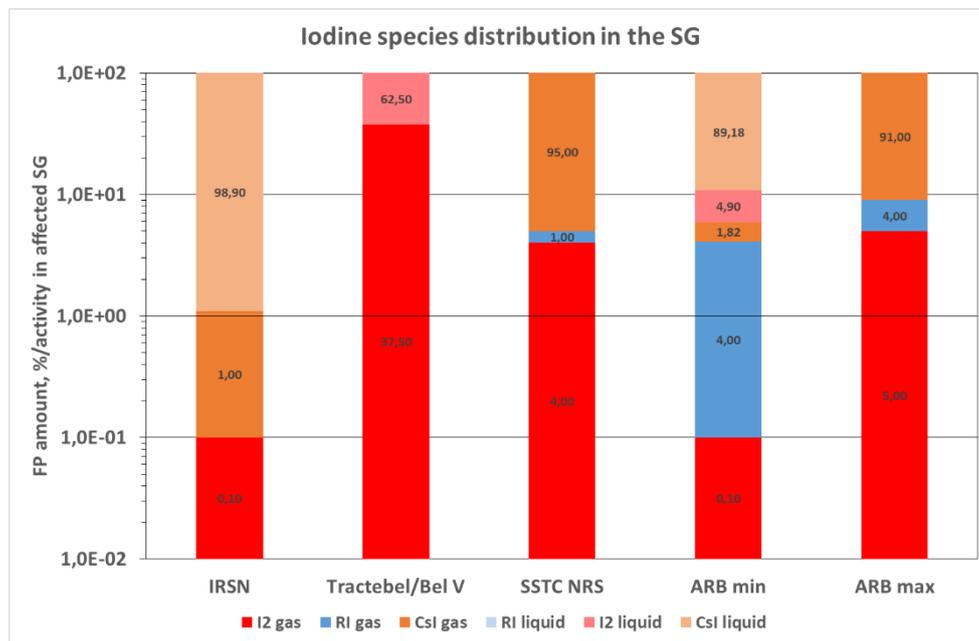


Figure 11. Relative amount of iodine containing species and their distribution in the gas and liquid phases of the SG

4.4 Secondary loop retention

In a SGTR accident scenario, the main FP release in the environment is assumed to occur through, for example, the suppression valves of the main steam lines. However, FP can be transported through the secondary loop into the condenser and the turbine building. All of the participants except Tractebel & Bel V do not consider any leak through the secondary loop to the turbine building or directly to the environment.

Tractebel & Bel V consider a release to the turbine building *via* the condenser, which originates from the degassing process (minor contribution to the total release). A decontamination factor considering the ion treatment of the blowdown system and a partitioning coefficient (typically a value of 44% removed from the condenser) is taken into account. Regarding the release directly to the environment, in addition to the aforementioned isenthalpic flashing and droplet atomisation, there is also the evaporation of the affected SG liquid phase, function of a partitioning coefficient (this partitioning coefficient depends on the liquid mass in the SG, if liquid mass > 1t, PC=100 and if liquid mass < 1t – dry-out, PC=1). The releases from the intact steam generators are also taken into account, even if their contribution to the total release is relatively small.

Figure 12 shows the iodine release into the environment with respect to the primary coolant activity released at the break. SSTC NRS and ARB (conservative hypothesis) have the most important source term (100%). The lowest source term is obtained by ARB for the “realistic” hypothesis (5.92%). IRSN and Tractebel/Bel V both present intermediate levels of releases for iodine (56.4% and 27.0% respectively). However, the Tractebel/Bel V source term is a low-boundary value, comprising the flashing, atomisation and partitioning phenomena taking into account the potential dry-out of the affected SG which could significantly increase the source term depending on the evolution of the thermal-hydraulic parameters. Regarding noble gases, since all the participants assume a 100% transfer to the steam phase of the SG, all of the noble gases released in the SG at the break will eventually be released in the environment.

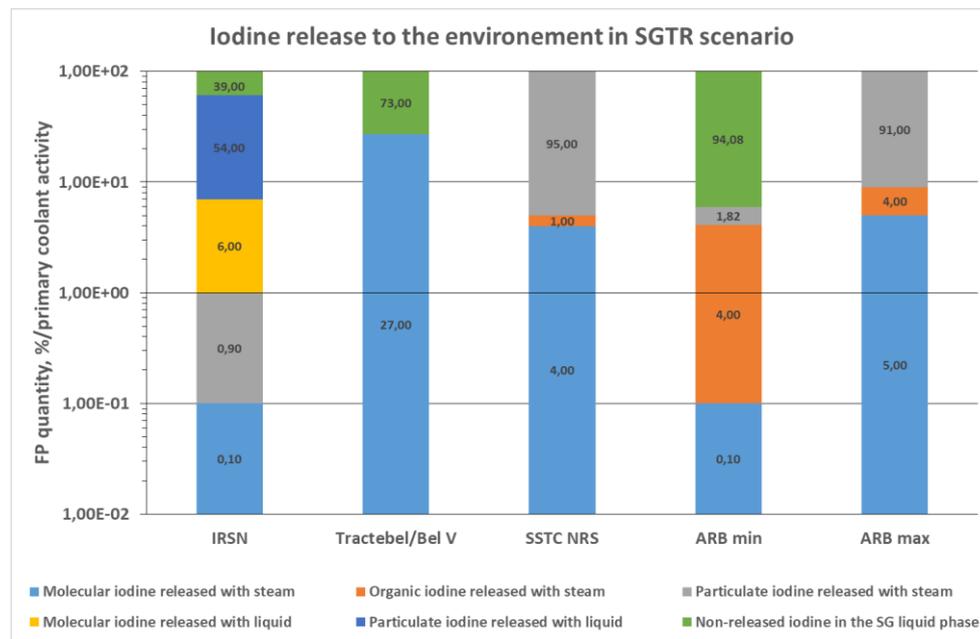


Figure 12. Iodine release in the environment in a SGTR scenario

5. Summary and conclusions

The evaluation of radiological consequences for DBA conditions are performed using very conservative deterministic assumptions, the majority being based on decoupled approaches. This does not allow an explicit quantification of gains, notably in terms of RC, of additional safety measures or devices, improved accident management strategies and advanced design. The present report synthesizes the different RC evaluation methodologies among the R2CA partners. The general conclusion is that despite sharing some assumptions, the approaches employed by different project partners may differ significantly.

For the LOCA scenario, the differences start at the isotopic inventory definition. Both deterministic and Monte-Carlo methods are used to establish the fuel composition. In terms of burn-up, realistic, maximum as well as average values are used. The assumed rate of failed fuel rods is also different among the participants, ranging from 33% to 100%. However there is an agreement regarding the core discretization: most methodologies consider several different types of similar fuel assembly based on irradiation and core management.

Concerning the elements volatility, the list of considered FP is different for each project partner. Also the definition of semi-volatile and low-volatile varies for several elements.

Regarding the release of FP, two approaches exist: the conservative 100% release, employed mainly by Tractebel and Bel V for noble gases and iodine, and a less conservative release model, which considers only the gap release. In terms of FP release to the containment, all the participants consider a 100% release of the FP the containment, and thus no RCS retention. However, the FP distribution between the liquid and gas phases of the containment is modelled following two approaches: a conservative 100% distribution of all FP into the gas phase and a more realistic distribution of elements between the two phases according to their chemistry. In terms of FP chemistry in the containment, three approaches are considered: time-dependent modelling of iodine chemistry (IRSN and LEI), instant modelling (ARB and SSTC NRS) and no modelling of iodine speciation (Tractebel and Bel V). The hypotheses regarding the impact of the Containment Spray System also vary: either the CSS operation is not considered at all or it is considered with varying degrees of efficiency.

All the methodologies consider unfiltered gas releases to the environment, with leak rates that are sensibly the same. Liquid leaks outside the containment are considered only by IRSN and Tractebel and

Bel V. There are differences concerning the filtered releases, specifically the efficiency of filtering systems towards the considered iodine species (molecular iodine, organic iodine and particulate iodine).

For the SGTR scenario there are also differences in the approaches for evaluating the RC. Tractebel and Bel V only consider ^{131}I for RC evaluation, whereas IRSN computes the equivalent ^{131}I activity, which is the weighted sum of the most important iodine isotopes present in the RCS. In addition IRSN, ARB and SSTC NRS consider the presence of other isotopes in the primary coolant, although the activity values vary. The iodine form in the RCS is also different: Tractebel and Bel V does not discriminate between iodine species, while the other methodologies consider molecular and particulate iodine. Moreover, ARB and SSTC NRS also consider organic iodine. For the distribution of iodine species between the liquid and gas phase of the steam generator, Tractebel and Bel V use a realistic approach based on phenomena such as isenthalpic flashing and droplet atomization, whereas the other participants consider partitioning coefficients with a varying degree of conservatism. Regarding the retention in the secondary loop, only Tractebel and Bel V consider any for iodine, while the other methodologies assume no iodine retention.

As a result, all the aforementioned differences in the hypotheses lead to different results regarding the FP release in the environment, respectively to different radiological consequences for the population. This in turn affects directly the decision making process in post-accident scenario.

References

1. Determination of the in-containment source term for a Large-Break Loss of Coolant Accident, Final report Contract B4-3070-97-810-MBL-C2; EUR 19841 EN, European Commission, April 2001.
2. W. Stephenson et al., Realistic methods for calculating the releases and consequences of a large LOCA; Final report Contract No. ETNU-0001/UK; EUR 14179 EN 1992.
3. Regulatory Guide 1.4, Assumptions used for evaluating the potential radiological consequences of a loss of coolant accident for pressurized water reactors. U.S. Atomic Energy Commission, June 1974.
4. Technical Report on the definition of the primary coolant source terms used in the different EPR designs for shielding, radiation zoning, DBA consequences, EPR Technical Report TR-EPRWG-03, Multinational Design Evaluation Programme, October 2015.
5. Regulatory Guide 1.183, Alternative radiological source terms for evaluating design basis accidents at nuclear power reactors. U.S. Nuclear Regulatory Commission, July 2000.
6. Regulatory Guide 1.195, Methods and assumptions for evaluating radiological consequences of design basis accidents at light-water nuclear power plants. U.S. Nuclear Regulatory Commission, May 2003.

Appendix A. Source Term evaluation methodology template

RC2A Task 2.1.1 (previous 1.1.1)

« Review of the RC evaluation methodologies »

Template for source term evaluation methodologies review

Task 2.1.1: Review of the source term evaluation methodologies

This task will be led by IRSN with a support of Bel-V for what concerns the PWR, of LEI for what concerns BWR, of SSTC_NRS and ARB for the VVER and VTT for the EPR. This action will notably benefit from the previous work of different groups such as the OECD Multinational Design Evaluation Programme (MDEP). Outcomes of this task will be shared to all partners together with the outcomes of task 1.1.2 and task 1.1.3 during a dedicated meeting. They will also be the first base to initiate the discussion with external groups such as WENRA, ENSREG, ETSON and the IAEA.

1. Instructions

The objective of this task is to build a state of the art of the source term evaluation methodologies between the partners, starting from the fuel to the environment release, for the considered accidents (LOCA and SGTR). For that purpose, please fill in the following tables with the current best practice of your institution, and return it to IRSN before January 31st.

Contact :

Jean-Jacques INGREMEAU, IRSN

jean-jacques.ingremeau@irsn.fr

For each table box, please fill it with, as a function of your practice:

- A constant value: **for example 10%**;
- A formula, describing the notations and the units: **for example Nikuradse equation** $K \sqrt{1 - \left(\frac{P_{atm}}{P}\right)^2} \sqrt{\frac{T}{M}}$, **K being a constant coming from the nuclear power plant data or experiments equal to XX%, P_{atm} the atmosphere pressure, P the pressure, T the temperature in Kelvin and M the molar mass (g/mol)**
- The name of the code you are using to evaluate this transfer, if adequate the module name, and if possible a typical value or range of this transfer in accidental conditions: **ASTEC code, module Sophaeros, 1-5%**.

In almost all tables, an example is given: “**Ex: 100%**”. The figures presented in these examples are only for illustration, and are not necessary realistic hypothesis.

For each transfer coefficient, the tables are function of the fission product type and form. The following groups are used and defined below:

- Noble Gas (NG): Xe and Kr
- Volatile Fission Product: For example I, Cs,
- Semi-volatile Fission Products: For example Ba, Ru
- Low Volatile Fission Products: For example Sm, Co
- Short live fission product: Fission products with half-life < 1 year
- Long live fission product: Fission products with half-life > 1 year
- Aerosols: small particles, being liquid or solid

Four main iodine species are taken into account in the following:

- Inorganic iodine (I₂);
- Metallic iodine aerosol (CsI);
- Organic iodine (RI);
- Iodine Oxides (IO_x).

In the case where you are not evaluating this transfer, or not able to answer, let it free.

In the case where your modelling can not be described in the following tables (for example you have several values as a function of other parameters), please adapt the table and inform IRSN (jean-jacques.ingremeau@irsn.fr).

If any question, ask IRSN (jean-jacques.ingremeau@irsn.fr).

2. About you

Table 16 : About you

First name	
Name	
Compagny	
E-mail address	
Reactor type of interest in RC2A project	

3. Isotopic inventory

Please fill in the table below the main hypothesis considered for the isotopic inventory.

Table 17 : Hypothesis for isotopic inventory evaluation

Code and method used	<i>Ex: Deterministic / Monte Carlo</i>
Discretisation	<i>Ex: One average fuel assembly / average of X different types of similar fuel assembly</i>
Burn-up considered	<i>Ex: 45 GWd/t / Average burn-up at end of cycle / Average burn-up at medium cycle / Maximum local burn-up / Realistic burn-up distribution for each assembly type at end of cycle</i>
Conditions	<i>Ex: Steady state at full nominal power / Power at end of stretch</i>

4. Source term methodology in LOCA conditions

4.1. Fuel release

4.1.1. Elements Volatility

Please fill in the table below the Fission Product (FP) elements considered and their volatility.

Table 18 : Elements and volatility

	Volatile FP	Semi-Volatile FP	Low Volatile FP
Elements	<i>Ex: I, Cs</i>		

4.1.2. Fuel to coolant

Please fill in the table below the considered fuel to coolant transfer in case of fuel failure, for LOCA conditions as a fraction of the inventory of the unsealed fuel.

Table 19 : Transfer from fuel to coolant

From fuel to coolant	Noble Gas		Volatile FP		Semi-Volatile FP		Low Volatile FP	
	Short live FP	Long Live FP	Short live FP	Long Live FP	Short live FP	Long Live FP	Short live FP	Long Live FP
Release fraction	<i>Ex: 5%</i>							

4.1.3. Fuel to steam (gas phase)

Please fill in the table below the considered fuel to the gas phase (steam, air, ...) transfer in case of dewatered fuel failure, as a fraction of the inventory of the unsealed fuel.

Table 20 : Transfer from fuel to the gas phase

From fuel failure to gas phase	Noble Gas		Volatile FP		Semi-Volatile FP		Low Volatile FP	
	Short live FP	Long Live FP	Short live FP	Long Live FP	Short live FP	Long Live FP	Short live FP	Long Live FP
Release fraction	<i>Ex: 5%</i>							

4.2. Releases in the containment

4.2.1. Reactor coolant system to containment release

Please fill in the table below the considered transfer to gas and liquid phase of the containment in case of LOCA, as a fraction of the amount of contaminated water released at the breach. The potential radioactive releases in the gas phase in the primary loop are supposed to be released directly in the gas phase of the containment.

Table 21 : Transfer from RCS Breach to the Containment

Transfer from the RCS breach to the containment	Noble Gas	Aerosols
Gas phase	<i>Ex: 100%</i>	
Liquid phase		

4.2.2. Iodine form release in the gas phase of the containment

Table 22 : Iodine form of the iodine release in the gas phase of the containment

Iodine Form	I ₂	CsI
If the breach is wet	<i>Ex: 3%</i>	
If the breach is dry		

4.3. Fission Product Behaviour in the Containment

Please fill in the tables below the considered phenomenon and the way to evaluate them, occurring for the Noble Gas, Iodines and Aerosols in the containment. The 4.3.1 part, describes the instantaneous phenomenon considered occurring at the release in the containment. The 4.3.2 describes the phenomenon occurring as a function of time. All the thermal-hydraulic parameters are supposed to be known (temperatures, pressure, steam condensing, steam production ...).

4.3.1. Instantaneous modelling

The instantaneous modelling of the phenomenon occurring in the containment are divided in three tables:

- A first one Table 23, for the phenomenon in the gas phase;
- A second one Table 24, for the phenomenon in the liquid phase;
- A third one Table 25, for the transfer between the liquid and the gas phase.

Table 23 : Instantaneous phenomenon in the gas phase of the containment

In the containment		Inorganic iodine (I ₂)	Metallic iodine aerosol (CsI)	Organic iodine (RI)	Iodine Oxides (IO _x)	Others aerosols
In the gas phase	Production				<i>Ex: 10% of I₂</i>	
	Consumption					

Table 24 : Instantaneous phenomenon in the liquid phase of the containment

In the containment		Inorganic iodine (I ₂)	Iodine ions (I ⁻)	Iodine oxide ions (IO ₃ ⁻)	Organic iodine (RI)	Others radionuclides
In the liquid phase	Production					
	Consumption					

Table 25 : Instantaneous transfer in the Containment between liquid and gas phase

In the containment	Noble Gas	Inorganic iodine (I ₂)	Metallic iodine aerosol (CsI)	Organic iodine (RI)	Iodine Oxides (IO _x)	Others radionuclides

Transfer from gas phase to liquid phase			<i>Ex: 50% (condensation)</i>			
Transfer from liquid phase to gas phase						

4.3.2. Time dependant modelling

The same decomposition than for the instantaneous modelling is proposed in the Table 26, Table 27 and Table 28 for time dependant phenomenon.

Table 26 : Time dependant phenomenon in the gas phase of the containment in the

In the containment		Noble Gas	Inorganic iodine (I ₂)	Metallic iodine aerosol (Csl)	Organic iodine (RI)	Iodine Oxides (IO _x)	Others aerosols
In the gas phase	Production						
	Consumption						

Table 27 : Time dependant phenomenon in the liquid phase of the containment

In the containment		Noble Gas	Inorganic iodine (I ₂)	Iodine ions (I ⁻)	Iodine oxide ions (IO ₃ ⁻)	Organic iodine (RI)	Others radionuclides
In the liquid phase	Production						
	Consumption						

Table 28 : Time dependant transfer in the Containment between liquid and gas phase

In the containment	Noble Gas	Inorganic iodine (I ₂)	Metallic iodine aerosol (Csl)	Organic iodine (RI)	Iodine Oxides (IO _x)	Others radionuclides
Transfer from gas phase to liquid phase						
Transfer from liquid phase to gas phase		<i>Ex: ASTEC code, module Sophaeros, ~1%/day</i>				

4.3.3. Containment Spray System impact

Please fill in the table below the considered efficiency of the containment spray system to transfer radionuclides from the gas phase to the liquid phase of the containment. Please describe in the box if the effect is instantaneous or considered to be time dependant.

Table 29 : Containment Spray System impact

	Noble Gas	Inorganic iodine (I ₂)	Organic iodine (RI)	Aerosols
Transfer from the gas phase to the liquid phase				

4.3.4. Decay and filiation

Are you taking into account these phenomena? Please fill the box with a "X".

Table 30 : Decay and filiation

	Yes	No
--	-----	----

Decay of fission products	<i>Ex: X</i>	
Filiation (production of others fission products due to parents decay)		

4.4. Containment releases

The containment releases are divided in unfiltered releases, due to containment leaks (liquid and gas), and filtered gas releases in a dedicated venting system. All the thermal-hydraulic parameters are supposed to be known (temperatures, pressure, venting system mass flow rate and period of operations).

4.4.1. Unfiltered releases

The unfiltered containment releases are divided in gas releases and liquid releases.

Table 31 : Unfiltered gas releases

Leakage rate	Noble Gas	Inorganic iodine (I ₂)	Organic iodine (RI)	Aerosols
Containment leakage	<i>Ex: Nikuradze equation</i> $K \sqrt{1 - \left(\frac{P_{atm}}{P}\right)^2} \sqrt{\frac{T}{M}}$ <small><i>K being a constant coming from the nuclear power plant data or experiments equal to XX%, P_{atm} the atmosphere pressure, P the pressure, T the temperature in Kelvin and M the molar mass (g/mol)</i></small>			
Third barrier extension leakage in adjacent buildings				
Adjacent buildings leakage	<i>Ex: 10%/day</i>			

Table 32 : Unfiltered liquid releases

Leakage rate	Liquid releases
Containment leakage	
Third barrier extension leakage in adjacent buildings	<i>Ex: 100L/h</i>

Table 33 : Transfer from liquid to gas releases outside of the containment

	Noble Gas	Inorganic iodine (I ₂)	Organic iodine (RI)	Metallic iodine aerosol (CsI)	Iodine Oxides (IO _x)	Others radionuclides
Transfer from liquid phase to gas phase outside of the containment	<i>Ex: 100%</i>					

4.4.2. Filtered releases

The retention factor is defined as the ratio between the volume activity before the filter and after the filter. For example, a retention factor of 1 means no filtration. A retention factor of 10 means the volume activity is divided by ten.

Table 34 : Filtered releases

Leakage rate	Noble Gas	Inorganic iodine (I ₂)	Organic iodine (RI)	Aerosols
Retention factor	<i>Ex: 1</i>			<i>Ex: 1000</i>

5. Source term methodology in SGTR conditions

5.1. Primary loop activity

Please fill the following table with the considered isotopes and their mass or volume activity in the primary loop during a Steam Generator Tube Rupture accident.

Table 35 : Primary loop activity

Primary loop activity	Mass or volume activity (precise in the unit)
⁸⁸ Kr	<i>Ex: 100GBq/t</i>
¹³⁵ Xe	
¹³³ Xe	
¹³¹ I	
¹³³ I	
¹³⁴ Cs	
¹³⁷ Cs	
⁵⁶ Co	
⁵⁸ Co	
⁶⁰ Co	
...	

5.1.1. Iodine form in the primary loop

Table 36 : Iodine form in the primary Loop

Iodine Form	Inorganic iodine (I ₂)	Metallic iodine aerosol (CsI)	Others
In the primary loop	<i>Ex: 0,1%</i>		

5.2. Transfer in the Steam Generator

Please fill in the table below the considered transfer to gas and liquid phase of the steam generator in case of SGTR, as a fraction of the amount of contaminated water released at the tube. All the thermal-hydraulic parameters are supposed to be known.

Table 37 : Transfer to gas phase of the Steam Generator

Transfer to gas phase of the Steam Generator	Noble Gas	Inorganic iodine (I ₂)	Aerosols
If the breach is located in water in the SG	<i>Ex: 100%</i>		
If the Breach is located in a mix Steam / Water (boiling water) in the SG			

If the Breach is located in a dry part of the SG			
--	--	--	--

Table 38 : Transfer to liquid phase of the Steam Generator

Transfer to liquid phase of the Steam Generator	Noble Gas	Inorganic Iodine (I ₂)	Aerosols
If the breach is located in water in the SG	<i>Ex: 0%</i>		
If the Breach is located in a mix Steam / Water (boiling water) in the SG			
If the Breach is located in a dry part of the SG			

5.3. Secondary loop retention

The liquid and steam amount rejected to the environment, due to suppression valves for example, are supposed to be known from the thermal-hydraulic evaluation. All the radionuclides contained in these volumes are directly taken into account in the source term.

The next table describe the others transfer that can occur in this situation.

Table 39 : Transfer from secondary loop to the turbine building and environment

Transfer		Noble Gas	Inorganic Iodine (I ₂)	Aerosols
Through the turbine building	From the gas phase of secondary loop to the turbine building	<i>Ex: 100%</i>		
	From the liquid phase of secondary loop to the turbine building			
	Turbine building leakage	<i>Ex: 10%/day</i>		
Directly in the environment	From the liquid phase of secondary loop to the environment			
	From the liquid phase of secondary loop to the environment			